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NUREG-1048
Supplement No. 6

85

# Safety Evaluation Report

## related to the operation of Hope Creek Generating Station

Docket No. 50-354

Public Service Electric and Gas Company
Atlantic City Electric Company

**U.S. Nuclear Regulatory Commission**

**Office of Nuclear Reactor Regulation**

July 1986

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Supplement No. 6

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## related to the operation of Hope Creek Generating Station

Docket No. 50-354

Public Service Electric and Gas Company
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## ABSTRACT

Supplement No. 6 to the Safety Evaluation Report on the application filed by Public Service Electric and Gas Company on its own behalf as co-owner and as agent for the other co-owner, the Atlantic City Electric Company, for a license to operate Hope Creek Generating Station has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located in Lower Alloways Creek Township in Salem County, New Jersey. This supplement reports the status of certain items that had not been resolved at the time of the publication of the Safety Evaluation Report. This supplement supports the issuance of a full-power license to operate Hope Creek Generating Station.




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# 1 INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

## 1.1 Introduction

In October 1984, the U.S. Nuclear Regulatory Commission (NRC) staff issued its Safety Evaluation Report (SER) (NUREG-1048) on the application filed by Public Service Electric and Gas Company (PSE&G) (hereinafter referred to as the licensee) on its own behalf as co-owner and as agent for the other co-owner, the Atlantic City Electric Company, for a license to operate the Hope Creek Generating Station (Docket No. 50-354). At that time, the staff identified items that were not yet resolved with the licensee. Supplement Nos. 1, 2, 3, 4, and 5 to the SER were issued in March 1985, August 1985, October 1985, December 1985, and April 1986, respectively. Supplement No. 5 was issued in support of the issuance of the low-power license. On April 11, 1986, this license, NPF-50, was issued to allow Hope Creek to operate at power levels not in excess of 5% of rated power. Except for one issue, SER Outstanding Issue 5 (solid-state logic modules), all SER confirmatory and outstanding (open) issues were resolved in Supplement Nos. 1 through 5. The purpose of this supplement to the SER is to provide the staff evaluation of the resolution of this open issue, to document Technical Specification changes, and to report on the modification of certain license conditions. This supplement supports the issuance of a full-power license for Hope Creek.

Each of the following sections or appendices of this SER supplement is numbered the same as the corresponding SER section or appendix that is being updated. Appendix A is a continuation of the chronology of the staff's actions related to the processing of the Hope Creek application and lists letters between the NRC staff and the licensee in chronological order. Appendix B is a list of references cited in this report.* Appendix C documents the resolution of Unresolved Safety Issue A-43, "Containment Emergency Sump Reliability." Appendix D is a list of acronyms used herein. Appendix E identifies principal contributors to this SER supplement. Appendix S contains the staff's review of the power ascension program acceleration. Appendix T documents the staff's review of certain Technical Specification changes requested by the licensee. Appendix U contains the staff's evaluation of the General Electric Company report, "Probability of Missile Generation in General Electric Nuclear Turbines."

Copies of this SER supplement are available for inspection at the NRC Public Document Room at 1717 H Street, N.W., Washington, D.C., and at the Pennsville Public Library, 190 South Broadway, Pennsville, New Jersey. They are also available for purchase from the sources indicated on the inside front cover of this report.

The NRC Project Manager assigned to the operating license application for Hope Creek is Mr. David H. Wagner. Mr. Wagner may be contacted by writing to

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*Availability of all material cited is described on the inside front cover of this report.



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## 1.7 Outstanding Issues

The staff identified certain outstanding issues in the SER that had not been resolved with the licensee. The status of these issues is listed in Table 1.1 and discussed further in the indicated sections of this report.

## 1.8 Confirmatory Issues

The staff identified confirmatory items in the SER that required additional information to confirm preliminary conclusions. The status of these items is listed in Table 1.2.

## 1.9 License Condition Items

There are certain issues for which a license condition may be desirable to ensure that staff requirements are met by a specified date (Table 1.3). These conditions will be in the form of a condition in the body of the operating license.



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Table 1.1 Outstanding issues (revised Table 1.1 from Supplement No. 5)

| Issue | Status | SER section(s) |
|---|---|---|
| (1) Riverborne missiles | Closed | 3.4.1 (Supplement 4) |
| (2) Equipment qualification | Closed | 3.10, 3.11 (also 3.10, Supplements 2 and 3) |
| (3) Preservice inspection program | Closed | 5.2.4, 6.6 (also Supplement 3) |
| (4) GDC 51 compliance | Closed | 6.2.7 (Supplement 2) |
| (5) Solid-state logic modules | Closed | 7.3.2.5 |
| (6) Postaccident monitoring instrumentation | Closed | 7.5.2.3 (Supplement 2) |
| (7) Minimum separation between non-Class 1E conduit and Class 1E cable trays | Closed | 8.3.3.3.3 (Supplement 3) |
| (8) Control of heavy loads | Closed | 9.1.5 (Supplement 1) |
| (9) Alternate and safe shutdown | Closed | 9.5.1.4 (also Supplement 2) |
| (10) Delivery of diesel generator fuel oil and lube oil | Closed | 9.5.4.2 (Supplement 1) |
| (11) Filling of key management positions | Closed | 13.1 (Supplement 5) |
| (12) Training program items | | |
| (a) Initial training programs | Closed | 13.2.1.1 (Supplement 2) |
| (b) Requalification training programs | Closed | 13.2.1.2 (Supplement 2) |
| (c) Replacement training programs | Closed | 13.2.1.3 (Supplement 2) |
| (d) TMI issues I.A.2.1, I.A.3.1, and II.B.4 | Closed | 13.2.1.4 (Supplement 2) |



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Table 1.1 (Continued)

| Issue | Status | SER section(s) |
|---|---|---|
| (e) Nonlicensed training programs | Closed | 13.2.2 (Supplement 2) |
| (13) Emergency dose assessment computer model | Closed | 13.3.2.9 (Supplement 5) |
| (14) Procedures generation package | Closed | 13.5.2 (Supplement 5) |
| (15) Human factors engineering | Closed | 18.1, 18.2 (Supplement 5) |



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Table 1.2 Confirmatory issues (revised Table 1.2 from Supplement No. 5)

| Issue | Status | SER section(s) |
|---|---|---|
| (1) Feedwater isolation check valve analysis | Closed | 3.6.2 (Supplement 3) |
| (2) Plant-unique analysis report | Closed | 3.9.3.1, (also Supplement 3) |
| (3) Inservice testing of pumps and valves | Closed | 3.9.6 (Supplement 4) |
| (4) Fuel assembly accelerations | Closed | 4.2 (Supplement 2) |
| (5) Fuel assembly liftoff | Closed | 4.2 (Supplement 2) |
| (6) Review of stress report | Closed | 5.2.1.1 (Supplement 3) |
| (7) Use of Code cases | Closed | 5.2.1.2 (Supplement 2) |
| (8) Reactor vessel studs and fasteners | Closed | 5.3.1.5 (Supplement 3) |
| (9) Containment depressurization analysis | Closed | 6.2.1.4 (Supplement 5) |
| (10) Reactor pressure vessel shield annulus analysis | Closed | 6.2.1.5 (Supplement 3) |
| (11) Drywell head region pressure response analysis | Closed | 6.2.1.5 (Supplement 3) |
| (12) Drywell-to-wetwell vacuum breaker loads | Closed | 6.2.1.7 (Supplement 3) |
| (13) Short-term feedwater system analysis | Closed | 6.2.3 (Supplement 3) |
| (14) Loss-of-coolant-accident analysis | Closed | 6.3.5, 15.9.3 (Supplement 2) |
| (15) Balance-of-plant testability analysis | Closed | 7.2.2.3 (Supplement 4) |
| (16) Instrumentation setpoints | Closed | 7.2.2.5 (Supplement 4) |



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Table 1.2 (Continued)

| Issue | Status | SER section(s) |
|---|---|---|
| (17) Isolation devices | Closed | 7.2.2.6 (Supplement 5) |
| (18) Regulatory Guide 1.75 | Closed | 7.2.2.7 (Supplement 5) |
| (19) Reactor mode switch | Closed | 7.2.2.9 (Supplement 3) |
| (20) Engineered safety features reset controls | Closed | 7.3.2.6 (Supplement 5) |
| (21) High pressure coolant injection initiation | Closed | 7.3.2.9 (Supplement 3) |
| (22) IE Bulletin 79-27 | Closed | 7.4.2.1 (Supplement 3) |
| (23) Bypassed and inoperable status indication | Closed | 7.5.2.4 (Supplement 4) |
| (24) Logic for high pressure coolant injection interlock circuitry | Closed | 7.6.2.1 (Supplement 4) |
| (25) End-of-cycle recirculation pump trip | Closed | 7.6.2.4 (Supplement 3) |
| (26) Multiple control system failures | Closed | 7.7.2.1 (Supplement 3) |
| (27) Relief function of safety/relief valves | Closed | 7.7.2.2 (Supplement 3) |
| (28) Main steam tunnel flooding analysis | Closed | 8.3.3.1.4 (Supplement 3) |
| (29) Cable tray separation testing | Closed | 8.3.3.3.2 (Supplement 3) |
| (30) Use of inverter as isolation device | Closed | 8.3.3.3.4 (Supplement 3) |
| (31) Core damage estimate procedure | Closed | 9.3.2 (Supplement 3) |
| (32) Continuous airborne particulate monitors | Closed | 12.3.4.2 (Supplement 3) |
| (33) Qualifications of senior radiation protection engineer | Closed | 12.5.1 (Supplement 2) |



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Table 1.2 (Continued)

| Issue | Status | SER section(s) |
|---|---|---|
| (34) Onsite instrument information | Closed | 12.5.2 (Supplement 3) |
| (35) Airborne iodine concentration instruments | Closed | 12.5.2 (Supplement 3) |
| (36) Emergency Plan items | Closed | 13.3 (Supplement 5) |
| (37) TMI Item II.K.3.18 | Closed | 15.9.3 (Supplement 2) |
| (38) Independent design verification | Closed | 17.5 (Supplement 5) |



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Table 1.3 License conditions (revised Table 1.3 from Supplement No. 5)

| License condition | Status | SER section |
|---|---|---|
| (1) Turbine system maintenance program | Removed | 3.5.1.3.3 |
| (2) NUREG-0803 implementation | Removed | 4.6 (Supplement 5) |
| (3) Inservice inspection | Revised | 5.2.4.3 and 6.6.3 (Supplement 5) |
| (4) Postaccident sampling system | Removed | 9.3.2 (Supplement 3) |
| (5) Solid waste process control program | Revised | 11.4 (Supplement 5) |
| (6) Partial feedwater heating | Revised | 15.1 |
| (7) Cask drop accident | Removed | 15.7.5 (Supplement 3) |
| (8) Inservice testing of pumps and valves | | 3.9.6 (Supplement 4) |
| (9) Environmental qualification | Removed | 3.11.5 |
| (10) Fire protection | Revised | 9.5.1 |
| (11) Emergency planning | | 13.3.3 (Supplement 5) |
| (12) Initial startup test program | | 14.2 (Supplement 5) |
| (13) Detailed control room design review | | 18.1 (Supplement 5) |
| (14) Fuel storage and handling | | 9.1 (Supplement 5) |
| (15) Safety parameter display system | | 18.2 (Supplement 5) |
| (16) Solid-state logic modules | | 7.3.2.5 |



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# 3 DESIGN CRITERIA FOR STRUCTURES, SYSTEMS, EQUIPMENT, AND COMPONENTS

## 3.5 Missile Protection

### 3.5.1 Missile Selection and Description

#### 3.5.1.3 Turbine Missiles

##### 3.5.1.3.3 Summary and Conclusions

License Condition 2.C.(3) of Facility Operating License NPF-50 for Hope Creek requires that PSE&G submit a turbine system maintenance program based on the turbine manufacturer's calculations of missile generation probabilities no later than April 11, 1989. In a letter dated July 7, 1986 (C. McNeill, PSE&G, to E. Adensam, NRC), the licensee submitted a turbine system maintenance program based on the proprietary General Electric Company (GE) report, "Probability of Missile Generation in General Electric Nuclear Turbines." The GE report was submitted by the licensee in a letter dated July 11, 1986 (C. McNeill, PSE&G, to E. Adensam, NRC).

The intent of the turbine system maintenance program is to provide early detection of cracking in the low-pressure turbine wheels, and thereby reduce the probability of turbine wheel failure. The maintenance program submitted by the licensee will include disassembly of the low-pressure turbines in stages over a 6-year period (during plant shutdowns) so that all low-pressure turbine wheels are inspected within the 6-year period. The program includes a complete inspection of all normally inaccessible parts such as couplings, coupling bolts, turbine shafts, low-pressure turbine buckets, and the low-pressure turbine wheels. The low-pressure turbine wheels will be subject to volumetric examination.

The staff has reviewed the licensee's turbine system maintenance program and concludes that the low-pressure turbine wheel inspection interval of 6 years is supported by the missile generation probabilities derived from the GE report, "Probability of Missile Generation in General Electric Nuclear Turbines." Furthermore, License Condition 2.C.(3) of the Hope Creek license (NPF-50) has been satisfied, and the staff recommends deletion of this license condition from the full-power license. (The staff's evaluation of the GE report submitted by the licensee's July 11, 1986, letter, is contained in Appendix U to this supplement.)

## 3.11 Environmental Qualification of Mechanical and Electrical Equipment

### 3.11.5 Conclusion

License Condition 2.C.(5)a of Facility Operating License NPF-50 for Hope Creek requires that, before startup following the first refueling outage, the qualified life of the electrical equipment under Purchase Order M-48 shall be recalculated on the basis of the actual temperatures monitored at the equipment locations during the first cycle of operation, with adequate consideration of margin.



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The staff recommended this license condition be included in the license because during the environmental qualification audit, it was noted that the licensee used the calculated average temperature for one Purchase Order M-48 item (primary containment instrument gas compressor).

In a letter dated June 2, 1986 (C. McNeill, PSE&G, to E. Adensam, NRC), the licensee indicated that it has reevaluated the issue and has revised qualified life values for electrical equipment purchased under Purchase Order M-48 for the maximum normal service temperature. The new values have been incorporated in the Hope Creek maintenance and surveillance program to reflect the revised replacement schedule for the components.

The staff has reviewed the information provided in the licensee's June 2, 1986, letter and concludes that the intent of the license condition has been met. Accordingly, the license condition can be deleted.

License Condition 2.C.(5)b of Facility Operating License NPF-50 for Hope Creek requires that, before initial criticality, the 53 Tobar Model 32, Series 2 transmitters included in the harsh environment qualification program be replaced with qualified Rosemount Model 1153 B transmitters. In a letter dated May 27, 1986 (C. McNeill, PSE&G, to E. Adensam, NRC), PSE&G confirmed that this had been done. Accordingly, License Condition 2.C.(5)b of the low-power license is satisfied and no longer required.



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# 7 INSTRUMENTATION AND CONTROLS

## 7.2 Reactor Protection (Trip) System

### 7.2.2 Specific Findings

#### 7.2.2.8 Anticipated Transients Without Scram

On February 25, 1983, both of the scram circuit breakers at Unit 1 of the Salem Nuclear Power Plant failed to open on an automatic reactor trip signal from the reactor protection system. This incident occurred during the plant startup, and the reactor was tripped manually by the operator about 30 sec after the initiation of the automatic trip signal. The failure of the circuit breakers has been determined to be related to the sticking of the undervoltage trip attachment. Before this incident, on February 22, 1983, at Unit 1 of the Salem Nuclear Power Plant, an automatic trip signal was generated based on steam generator low-low level during plant startup. In this case, the reactor was tripped manually by the operator almost coincidentally with the automatic trip. Following these incidents, on February 28, 1983, the NRC Executive Director for Operations directed the NRC staff to investigate and report on the generic implications of these occurrences at Unit 1 of the Salem Nuclear Power Plant. The results of the staff's inquiry into the generic implications of the incidents at the Salem unit are reported in NUREG-1000, "Generic Implications of ATWS [Anticipated Transient Without Scram] Events at the Salem Nuclear Power Plant." As a result of this investigation, the Commission (NRC) requested (by Generic Letter 83-28 dated July 8, 1983) all licensees of operating reactors, applicants for an operating license, and holders of construction permits to respond to certain generic concerns.

The licensee responded to these concerns by letters dated March 30 and December 17, 1984; February 28, May 21, July 15, August 7, and October 30, 1985; and January 24, 1986.

In a letter dated June 9, 1986 (E. Adensam, NRC, to C. McNeill, PSE&G), the status of the staff's review of these responses was sent to the licensee. Table 7.1 presents the Hope Creek status of all Generic Letter 83-28 issues applicable to boiling-water reactors. The licensee's responses to all items, except for Items 2.1, 2.2, 4.5.2, and 4.5.3, have been reviewed and found acceptable. As noted in Table 7.1, the review for Items 2.1, 2.2, 4.5.2, and 4.5.3 is ongoing. These four items are primarily administrative in nature and do not affect the plant design (or the implementation of that design). The review status of these items is similar to that of many operating plants. Therefore, the staff does not consider the completion of these items a prerequisite for authorizing full-power operation. Should the staff's review conclude that additional actions by the licensee are required to be in compliance with Generic Letter 83-28, the licensee will be notified.



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## 7.3 Engineered Safety Feature Systems

### 7.3.2 Specific Findings

#### 7.3.2.5 Solid-State Logic Modules

With regard to logic functional testing of Bailey solid-state logic modules (SSLMs), the staff recommended in Supplement No. 5 that the licensee review the existing test method (performed every 18 months under cold conditions) to see if the test, with necessary modifications, can be used at power (without challenging plant systems and creating undue competing risks) to demonstrate SSLM reliability. Additionally, the licensee was asked to investigate other methods by which testability frequency can be increased.

In a letter dated May 23, 1986 (C. McNeill, PSE&G, to E. Adensam, NRC), the licensee provided (1) a listing of Class 1E Bailey 862 SSLMs, which perform logic functions, that are operated or surveilled in the performance of Hope Creek Technical Specification surveillances, (2) a description of Technical Specification surveillance testing performed on the Bailey 862 SSLMs, and (3) a description of four options (described below) available to perform in situ, at-power surveillance testing of the surveilled Bailey 862 SSLMs identified in Item (1) above and the effect associated with performing each of the four options. Also contained in the May 23, 1986, letter was a proposal from the licensee to develop a program by which reliability data associated with the Bailey 862 SSLMs will be gathered to demonstrate the reliability of the Bailey 862 SSLMs.

(1) Option 1

Option 1 would maintain the current Hope Creek configuration and, where possible, utilize existing procedures, or portions of these procedures, that were developed to comply with Technical Specification surveillance requirements. These surveillance tests would be performed on a more frequent basis than currently required (once every 18 months) by the Technical Specifications.

The effects of performing Option 1 would include

(a) additional person-hours and cost to perform the surveillance testing

(b) potential to enter limiting conditions for operation for those systems rendered inoperable as a result of testing at power

(c) utilization of jumpers and lifted leads to prevent undesired isolations/actuations and to simulate various plant conditions

(d) disabling Bailey 862 SSLMs logic outputs, thus disabling equipment required to be available to operate or possibly initiate safety functions or permissives on equipment already in operation

(e) challenging the ability to control and cool down the reactor in the event of an actual loss-of-coolant accident (LOCA) or loss of power



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(f) negating the purpose of performing logic functional testing because portions of the decision logic would be off line

(2) Option 2

Option 2 would maintain the current Hope Creek configuration and develop new surveillance test procedures to allow in situ testing while at power. The specific intent of these new procedures would be to verify functionally the SSLMs used within a system.

The effects of performing Option 2 would include

(a) additional person-hours and extensive expense associated with developing unique procedures for in situ testing of logic modules

(b) potential to enter limiting conditions for operation for those systems rendered inoperable as a result of testing at power

(c) utilization of jumpers and lifted leads to prevent undesired isolations/actuations and to simulate various plant conditions

(d) disabling Bailey 862 SSLM logic outputs, thus disabling equipment required to be available to operate or possibly initiate safety functions or permissives on equipment already in operation.

(e) challenging the ability to control and cool down the reactor in the event of an actual LOCA or loss of power

(f) negating the purpose of performing logic functional testing because portions of the decision logic would be off line

(3) Option 3

Option 3 would maintain the current Hope Creek configuration, develop new surveillance test procedures to allow in situ testing while at power, and test 100% of a total test population of 577 Bailey 862 SSLMs during each cycle of operation, as outlined in letters dated February 14 and 24, 1986 (C. McNeill, PSE&G, to E. Adensam, NRC).

The effects of performing Option 3 would include

(a) additional person-hours and expense to perform the testing

(b) disabling Bailey SSLM logic outputs, thus disabling equipment required to be available to operate or possibly initiate safety functions or permissives on equipment already in operation

(c) challenging the ability to control and cool down the reactor in the event of an actual LOCA or loss of power

(d) negating the purpose of performing logic functional testing because portions of the decision logic would be off line.



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(4) Option 4

Option 4 would maintain the current Hope Creek configuration and testing in accordance with current Technical Specification requirements. There are no effects or operational perturbations associated with performing Option 4.

On the basis of its review of the above information, the staff has concluded that the tests to be performed under Options 1, 2, or 3 described above will result in disabling of equipment required to be available to operate or possibly will inhibit safety function or permissives on equipment already in operation. With such a plant status at full power, if an emergency condition (LOCA or loss of offsite power) should occur, the plant's ability to cope with such a condition could be challenged. Recognizing this limitation in on-line testing, Regulatory Guide 1.22 exempts on-line testing of those systems/components whose testing at power adversely affects the safety or operability of the plant and allows them to be tested while the plant is shut down.

The licensee has stated that the percentage of SSLMs at Hope Creek that are tested monthly, quarterly, and semiannually is 21%, 36%, and 0.72%, respectively. The licensee has also indicated that the Bailey 862 SSLMs in operation at Colorado Ute, Craig Station, Unit 3, for the past 36 months had not experienced any failures.

Solid-state components and circuits are being increasingly used in safety-related systems in nuclear power plants. On the basis of the current operating experience, the staff believes that the solid-state systems in general are performing acceptably. Because the Bailey SSLMs have not been used in nuclear power plant application, it is the staff's determination that the licensee for Hope Creek should demonstrate the reliability of these modules.

On the basis of the above, the staff has concluded that the licensee should adopt Option 4 for SSLMs at Hope Creek.

In letters dated June 13 and 24, 1986 (C. McNeil, PSE&G, to E. Adensam, NRC), the licensee provided details of its reliability program to demonstrate the reliability of the Bailey 862 SSLMs. The reliability program would consist of the following three subprograms:

(1) The first subprogram would be an in-plant reliability program to monitor the performance of the Bailey 862 SSLMs installed at Hope Creek. This subprogram would obtain reliability data, failure characteristic information, and root cause of failure of both safety-related and non-safety-related Bailey 862 SSLMs for a period of at least 18 months. This will indicate the actual in-plant performance of the Bailey 862 SSLMs.

(2) The second subprogram would consist of an accredited laboratory performing physical testing of a statistical sample of the Bailey 862 SSLMs. These tests would simulate inputs and outputs as near to plant conditions as possible. The tests would include the effects of aging. The test results would be analyzed to verify that the modules will perform their intended safety functions under service conditions. The laboratory would generate a report to provide the results of the reliability testing and reliability analysis.



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(3) The third subprogram would be the collection of reliability data by Bailey Controls over a period of at least 18 months from other industrial installations of Bailey 862 SSLMs to provide an additional overall reliability basis on the modules.

The reliability program will have specific recommendations on qualified life, surveillance requirements, testing frequency, identification of specific parameters for degradation monitoring, preventive maintenance requirements, limiting conditions for operation, and any corrective actions.

Before the end of the first refueling outage, the licensee should provide an analysis of the results of the reliability program to demonstrate that the SSLMs can perform their safety functions in a reliable manner.

The licensee will submit the details of the reliability test program for staff review by August 15, 1986. The licensee in a letter dated June 24, 1986, has committed to submit for staff review the results of the above-stated reliability program before the end of the first refueling outage. The Hope Creek full-power license will be so conditioned.

On the basis of its review of the technical approach, methodology, acceptance criteria, and quality assurance procedures, the staff has concluded that the licensee's proposed reliability program is a viable approach to demonstrate the reliability of SSLMs at Hope Creek.

Because of the generic backfit implications of a reinterpretation of General Design Criterion 21 (Appendix A to 10 CFR 50) regarding testability of protection systems at power, the staff has categorized the on-line testability of protection systems as Generic Issue 120. Any requirements that emanate from the resolution of this generic issue will be applied to the protection system at Hope Creek.

Because Bailey 862 SSLMs at Colorado Ute, Craig Station, Unit 3, had no observed failures in the past 36 months of operation, a certain percentage of SSLMs at Hope Creek will be periodically exercised during power operation, and the surveillance requirements of the SSLMs are in accordance with the Standard Technical Specifications, the staff concludes that there is reasonable assurance that the SSLMs at Hope Creek will perform the required safety functions during the first cycle of operation.

The staff recommends that the following license condition be added to the full-power license:

> PSE&G shall implement a reliability program, to demonstrate solid state logic module reliability, as described in its letters dated June 13 and 24, 1986. The results of the reliability program shall be submitted to the staff prior to the end of the first refueling outage.

Table 7.1 Generic Letter 83-28 review for Hope Creek

| Item | Status | Date of letter transmitting safety evaluation to licensee |
|---|---|---|
| 1.1 | Review is complete. | January 22, 1986 |
| 1.2 | Review is complete. | February 3, 1986 |
| 2.1 | Review is ongoing. | |
| 2.2 | Review is ongoing. | |
| 3.1.1 | Review is complete. | June 25, 1986 |
| 3.1.2 | Review is complete. | June 25, 1986 |
| 3.1.3 | Review is complete. | January 22, 1986 |
| 3.2.1 | Review is complete. | June 25, 1986 |
| 3.2.2 | Review is complete. | June 25, 1986 |
| 3.2.3 | Review is complete. | January 22, 1986 |
| 4.5.1 | Review is complete. | June 25, 1986 |
| 4.5.2 | Review is ongoing. | |
| 4.5.3 | Review is ongoing. | |



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# 9 AUXILIARY SYSTEMS

## 9.5 Other Auxiliary Systems

### 9.5.1 Fire Protection

In a letter dated May 13, 1986 (C. McNeill, PSE&G, to E. Adensam, NRC), the licensee requested the deletion of Fire Protection Program (FPP) elements from the Hope Creek Generating Station Technical Specifications when the full-power license is issued. The FPP requirements defined in the Technical Specifications would be embodied within a periodic testing and surveillance program at Hope Creek and entail a shifting of testing requirements from surveillance procedures to periodic test procedures. In preparing the request, the licensee used the guidance contained in Generic Letter 86-10, "Implementation of Fire Protection Requirements."

In response to the licensee's request, the staff performed an inspection at the Hope Creek site to verify that the Technical Specification surveillances are incorporated into station procedures. The results of this inspection are reported in NRC Office of Inspection and Enforcement (IE) Inspection Report 50-354/86-29, dated June 25, 1986. In summary the staff found:

(1) The existing Final Safety Analysis Report (FSAR) sections of the FPP and the licensee's May 13, 1986, request, satisfy the guidance in Generic Letter 86-10 for incorporating the FPP into the FSAR.

(2) The deletion of the Technical Specification sections is in accordance with the guidance in Generic Letter 86-10.

(3) The existing fire protection Technical Specification requirements are incorporated into equivalent plant procedures, and equivalent administrative controls exist to control these activities.

(4) Adequate administrative controls exist to determine if a proposed FPP change would adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

The staff has reviewed the licensee's May 13, 1986, request and concludes that the incorporation of the Technical Specification surveillance procedures into periodic test procedures will result in an equivalent level of fire protection as would have resulted had the FPP elements remained in the Technical Specifications. Therefore, the licensee's request is acceptable.

In the Hope Creek low-power license, NPF-50, the fire protection license condition, 2.C.(8), stated, in part,

> PSE&G shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility through Amendment No. 13 and as



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> described in submittals dated August 12 and 27, November 21 and 29, December 3 and 16, 1985, January 3 and April 8, 1986, and as approved in the SER dated October 1984 (and Supplements 1 through 5)....

This license condition must be revised to incorporate the latest changes to the FPP and reference the licensee's May 13, 1986, request. FSAR Amendments 14 and 15 incorporated the submittals referenced in the license condition. Accordingly, these submittal dates may be replaced with the proper incorporation of Amendments 14 and 15 into the license condition. Additionally, the license condition is revised to reflect the licensee's commitments made in the May 13, 1986, request to delete FPP elements from the Technical Specifications. Accordingly, the new fire protection license condition reads as follows:

> PSE&G shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility through Amendment No. 15 and as described in its submittal dated May 13, 1986, and as approved in the SER dated October 1984 (and Supplements 1 through 6) subject to the following provision:
>
> > The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.



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## 14 INITIAL TEST PROGRAM

### 14.2 Initial Plant Test Program - Final Safety Analysis Report

At the request of the licensee, a meeting was held on August 1, 1985, to discuss the proposed acceleration of the Hope Creek power ascension program (PAP). (The meeting summary is dated August 19, 1985.) At the meeting, the licensee identified five categories of methods by which the PAP could be accelerated:

(1) Replace testing with Technical Specification surveillances.
(2) Delete nonessential testing.
(3) Simplify certain tests.
(4) Replace tests with data from other tests.
(5) Delete certain Regulatory Guide 1.68 testing.

In letters dated August 21, 1985 (R. Mittl, PSE&G, to W. Butler, NRC), September 20, 23, and 30, 1985 (R. Mittl, PSE&G, to W. Butler, NRC), October 4 and 17, 1985 (R. Mittl, PSE&G, to W. Butler, NRC), November 6, 1985 (R. Mittl, PSE&G, to W. Butler, NRC), and December 9, 1985 (C. McNeill, PSE&G, to E. Adensam, NRC), the licensee submitted PAP test modifications for NRC review. In total, modifications for 26 PAP tests were submitted. In letters dated January 22, February 4, and March 20, 1986 (R. Bernero, NRC, to C. McNeill, PSE&G), the staff provided a safety evaluation for each proposed test modification. These letters are included as Appendix S to this supplement.



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# 15 SAFETY ANALYSIS

## 15.1 Decrease in Core Coolant Temperature

By letter dated May 8, 1986, the licensee requested that the license condition, "Partial Feedwater Heating," be changed and that this change be incorporated into the full-power license. The license condition currently reads:

> PSE&G shall not operate the facility (other than for normal start-up or shutdown) with the feedwater inlet temperature less than 424.5°F.

The licensee informed the staff that, during normal operation at 75% power, the feedwater temperature is estimated to be 398°F. In this particular case, the licensee would be in violation of the license condition as it is currently written. Therefore, the licensee requests a revised license condition for the full-power license.

The staff understands that reduced feedwater temperature (RFT) operation is possible during normal operation. Feedwater temperature normally drops with decreasing power level while all feedwater heaters are in service and could be reduced during maintenance on the feedwater system.

The staff has reviewed analyses performed for a BWR/6 using staff-approved models to justify steady-state operation with RFTs. Temperatures ranging from 420°F to 250°F were considered. The anticipated operating transients were reevaluated to determine the required operating limit minimum critical power ratio (MCPR) limits. The operating limit MCPR values were found to increase by 0.01 for operation between feedwater temperatures of 370°F and 320°F and by 0.03 for feedwater temperatures between 320°F and 250°F.

The staff has reviewed the application of these findings to Hope Creek operations. Although plant-specific RFT analyses were not performed for Hope Creek, the staff believes operation with RFT, for the first cycle only, would be acceptable, provided that the Hope Creek Technical Specification operating limit MCPR is increased by 0.03 for a feedwater temperature between 400°F and 320°F and by 0.06 for a feedwater temperature between 320°F and 250°F. The additional margin is provided to account for possible differences in transient response between Hope Creek, a BWR/4, and the analyzed BWR/6.

The staff believes that in lieu of the license condition as currently written, the following license condition will permit greater operational flexibility while ensuring operation in analyzed regions:

> The facility shall not be operated with reduced feedwater temperature for the purpose of extending the normal fuel cycle. After the first operating cycle, steady state operation with reduced feedwater temperature (relative to the Final Safety Analysis Report analysis value) during the normal fuel cycle shall be prohibited until plant-specific analyses justifying such operation are provided by PSE&G and approved by the staff.



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In a letter dated June 23, 1986 (E. Adensam, NRC, to C. McNeill, PSE&G), the staff proposed this license condition to the licensee and requested that the licensee notify the staff if the proposed license condition and MCPR modifications were acceptable. In a letter dated July 7, 1986 (C. McNeill, PSE&G, to E. Adensam, NRC), the licensee informed the staff that the proposed license condition was acceptable, with certain clarifications. Additionally, the licensee proposed a revision to Technical Specification 3.2.3 to incorporate the MCPR modifications. This proposed Technical Specification revision included the addition of a graph of feedwater temperature versus power level to Technical Specification 3.2.3. The staff concluded that the addition of this graph to the Technical Specifications was not acceptable, since the graph had not been previously reviewed and approved by the staff.

On the basis of discussions with the licensee, the licensee and staff both agree that the following license condition, accompanied by an MCPR modification table added to the Technical Specifications, will meet the intent of the staff's June 23, 1986, letter to the licensee. Accordingly, the staff recommends that license condition, "Partial Feedwater Heating," be modified to state:

> The facility shall not be operated with reduced feedwater temperature for the purpose of extending the normal fuel cycle. After the first operating cycle, the facility shall not be operated with a feedwater heating capacity that would result in a rated thermal power feedwater temperature less than 400°F unless analyses supporting such operation are submitted by the licensee and approved by the staff.

## 16 TECHNICAL SPECIFICATIONS

The Technical Specifications in a license define certain features, characteristics, and conditions governing the operation of the facility that cannot be changed without prior approval of the NRC staff. The Hope Creek Technical Specifications are included as Appendix A to the Hope Creek license. Included in the Technical Specifications are sections covering definitions, safety limits, limiting safety system settings, limiting conditions for operation, surveillance requirements, design features, and administrative controls.

In letters dated May 13 and 30, June 4 and 13, and July 7 and 24, 1986 (C. McNeill, PSE&G, to E. Adensam, NRC), the licensee requested certain changes to the Technical Specifications issued as part of the Hope Creek license. The licensee requested these changes as a result of low-power operating experience to date. The requested changes and their status are identified in Table 16.1.

Table 16.1 Technical Specification changes

| Date of PSE&G letter request | Change description* | Status |
|---|---|---|
| May 13, 1986 | The licensee requested deletion of fire protection program elements from the Technical Specifications, in accordance with guidance contained in Generic Letter 86-10. | See Section 9.5.1 of this supplement. This change required modification of a license condition. |
| May 30, 1986 | The licensee requested changes to the Technical Specifications to permit operation with one recirculation loop out of service. | Review will be completed after full-power license is issued. |
| June 4, 1986 | The licensee requested changes to the Technical Specifications to permit operation with drywell and suppression chamber purge system valves under administrative control. | Review will be completed after full-power license is issued. |
| June 13, 1986 | The licensee requested changes to 13 Technical Specification sections based on operational experience achieved since issuance of the low-power license. These changes are: | |
| | (1) The licensee requested changes to Technical Specification 3/4.8.2 permitting one channel of dc power to be inoperable for 8 hours, instead of 2 hours, before requiring plant shutdown. | Review will be completed after full-power license is issued. |
| | (2) The licensee requested changes to Technical Specifications 4.6.5.3 and 4.7.2 to state that the filtration, recirculation, and ventilation system (FRVS) and control room emergency filtration system (CREFS) heaters will be on rather than just operable for the required surveillance. | See Appendix T, Item (1), of this supplement. |

*For a more complete discussion, see the referenced letter.



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Table 16.1 (Continued)

| Date of PSE&G letter request | Change description* | Status |
|---|---|---|
| June 13, 1986 | (3) The licensee requested changes to the Technical Specification index to enhance the index and make the Technical Specifications easier to use. | Changes are editorial and have been incorporated into the Technical Specifications. |
| | (4) The licensee requested changes to Technical Specification 3.8.1.1 to permit 24 hours, rather than 2 hours, to shut down the plant if two diesel generators are inoperable. | Review will be completed after full-power license is issued. |
| | (5) The licensee requested a change to Technical Specification 6.9.1.8 revising the address to which routine reports of operating statistics and shutdown experience are sent. | Change is editorial and has been incorporated into the Technical Specifications. |
| | (6) The licensee requested changes to Technical Specification Tables 2.2.1-1, 3.3.1-1, 3.3.1-2, and 4.3.1.1-1 to correct typographical errors. | Changes are editorial and have been incorporated into the Technical Specifications. |
| | (7) The licensee requested changes to Technical Specification 3.9.2 deleting the requirement that thermal power be less than 1% when one of three required source range monitors may be retracted. | Changes are editoral and have been incorporated into the Technical Specifications. |
| | (8) The licensee requested changes to Technical Specification 4.4.6.1.4 allowing the reactor vessel flange and head flange temperature to be reduced. | See Appendix T, Item (5), of this supplement. |

*For a more complete discussion, see the referenced letter.



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Table 16.1 (Continued)

| Date of PSE&G letter request | Change description* | Status |
|---|---|---|
| June 13, 1986 | (9) The licensee requested changes to Technical Specification Tables 3.3.3-1, 3.3.7.5-1, and 4.3.7.5-1 and Section 3.11.3 to delete various provisions that allowed completion of preoperational tests after initial fuel load but before 5% power is exceeded. | Changes are editorial and have been incorporated into the Technical Specifications. |
| | (10) The licensee requested changes to Technical Specification Table 3.6.3-1 to delete operability testing of excess flow check valve BB-XV-3649. | See Appendix T, Item (4), of this supplement. |
| | (11) The licensee requested changes to Technical Specification Table 3.6.3-1 to correct the identification of the Remote Manual Isolation Valves Group 26 - RHR Suppression Pool Return Valves as Other Primary Containment Isolation Valves Group 39 - RHR System, since these valves are manual, not remote manual valves. | Changes are editorial and have been incorporated into the Technical Specifications. |
| | (12) The licensee requested changes to Technical Specifications 4.6.5.3.e.3 and 4.7.2.e.4 to clarify the method of verifying humidity control instruments are operable by using a channel calibration for the FRVS and CREFS and to change the FRVS heater dissipation from 100 ± 5 kW to 100 ± 10 kW. | See Appendix T, Item (2), of this supplement. |
| | (13) The licensee requested changes to Technical Specification Table 3.3.7.1-1 to delete Provision (b), which requires that the offgas post-treatment radiation monitor be operable. | See Appendix T, Item (3), of this supplement. |

*For a more complete discussion, see the referenced letter.



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Table 16.1 (Continued)

| Date of PSE&G letter request | Change description* | Status |
|---|---|---|
| July 7, 1986 | The licensee requested changes to Technical Specification 3.2.3 in response to the staff's June 23, 1986, letter. | See Section 15.1 of this supplement. This change required modification of a license condition. |
| July 24, 1986 | The licensee requested changes to two Technical Specification sections based on operational experience achieved since issuance of the low-power license. These changes are: | |
| | (1) The licensee requested a change to Technical Specification 4.8.1.1.2.f.1.b changing the maximum Saybolt Second Universal (SSU) viscosity of new diesel generator fuel oil from 45 SSU to 40.1 SSU. | Change is editorial and has been incorporated into the Technical Specification. |
| | (2) The licensee requested that a Section 6.9.3 be added to the Technical Specifications which would require that violations of the Fire Protection Program be reported via a Licensee Event Report within 30 days. | The requested addition has been incorporated into the Technical Specifications. |

*For a more complete discussion, see the referenced letter.



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APPENDIX A

CONTINUATION OF CHRONOLOGY

| | |
|---|---|
| August 1, 1985 | Meeting between applicant and staff to discuss the proposed compression of the Hope Creek power ascension program. |
| March 20, 1986 | Letter to applicant forwarding staff's safety evaluation on the last set of power ascension program test modifications. |
| April 8, 1986 | Letter from applicant documenting a teleconference with the staff concerning the effect of a fire on high-pressure/ low-pressure interfaces. |
| April 10, 1986 | Letter from applicant requesting withdrawal of April 8, 1986, letter regarding clarification/definition of initial criticality and applicability to plant. |
| April 11, 1986 | Letter from applicant responding to NRC letter of March 14, 1986, regarding violations in Inspection Report 50-354/86-03. Revision 1 to Test Procedure TE-SU.ZZ-041 (Q) issued on March 8, 1986, will properly address identified deficiencies. |
| April 11, 1986 | Letter to applicant forwarding License NPF-50, environmental protection plan, Amendment 11 to Indemnity Agreement B-74, Supplement No. 5, and Federal Register notice of issuance. |
| April 11, 1986 | License NPF-50 issued for Hope Creek Generating Station. |
| April 14, 1986 | Letter from licensee forwarding 1985 Annual Report for PSE&G, Atlantic City Electric Co., Delmarva Power and Light Co., and Philadelphia Electric Co. |
| April 14, 1986 | Letter to licensee forwarding Inspection Report 50-354/ 86-14. Utility-sponsored SAFETEAM program satisfactorily identified and resolved employee concerns during final stages of plant construction and preoperational testing. |
| April 16, 1986 | Letter to licensee forwarding final Systematic Assessment of Licensee Performance Report 50-354/85-99 for November 1984-October 1985, per February 27, 1986, meeting and review of March 17, 1986, comments. |
| April 16, 1986 | Letter to licensee expressing appreciation for assistance in training effort during site visit on March 19, 1986, while preparing to load fuel in Unit 1. |
| April 17, 1986 | Letter to licensee forwarding Safety Inspection Report 50-354/86-21. |



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| April 18, 1986 | Letter to licensee expressing appreciation for recipient and M. Headrick participation in NRC Operator Licensing Subject Matter Expert Panel meetings during February and March 1986. |
| April 23, 1986 | Letter to licensee forwarding Safety Inspection Report 50-354/86-19. |
| April 23, 1986 | Letter to licensee forwarding Safety Inspection Report 50-354/86-18. |
| April 24, 1986 | Generic Letter 86-10 to all power reactor licensees and applicants for power reactor licenses regarding implementation of fire protection requirements. |
| April 28, 1986 | Letter to licensee forwarding initial Operating License (OL) Review Report 50-354/86-25. |
| April 28, 1986 | Letter to licensee requesting results of investigation regarding potential improprieties of quality control supervisors and proposed corrective actions within 30 days. |
| May 2, 1986 | Letter to licensee forwarding Examination Report 50-354/86-16 regarding examination administered during week of February 24, 1986. |
| May 8, 1986 | Letter from licensee forwarding application for amendment to License NPF-50, revising License Condition 2.C.12 regarding partial feedwater heating. Facility will not be operated with partial feedwater heating for purpose of extending normal fuel cycle without prior written staff consent. |
| May 9, 1986 | Letter to licensee acknowledging receipt of April 11, 1986, letter informing staff of steps taken to correct violations noted in Inspection Report 50-354/86-03. Corrective actions documented in Inspection Reports 50-354/86-22 and 50-354/86-23. |
| May 12, 1986 | Letter from licensee forwarding Licensee Event Reports (LERs) 86-002-00 and 86-003-00. |
| May 13, 1986 | Letter from licensee informing staff that, as of May 3, 1986, offgas system was successfully preoperationally tested, allowing tensioning of reactor pressure vessel head closure bolts in support of low-power testing activities per Supplement No. 5 and Appendix R. |
| May 13, 1986 | Letter from licensee discussing current program for engineering expertise on shift, shift technical advisor degree requirement, and proposed modifications to Final Safety Analysis Report (FSAR) regarding engineering expertise on shift and licensed operator staffing, per Generic Letter 86-04 and NUREG-0737. |



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| May 13, 1986 | Letter from licensee requesting deletion of fire protection program elements from Technical Specifications when full-power operating license is issued. |
| May 14, 1986 | Letter to licensee requesting results of investigation and proposed corrective action within 30 days regarding quality concerns with electrical supports at facility listed in April 29, 1986, anonymous letter. |
| May 20, 1986 | Letter to licensee forwarding Safety Inspection Report 50-354/86-20 and notice of violation and acknowledging receipt of April 2, 1986, response to violations cited in Inspection Report 50-354/86-06. |
| May 22, 1986 | Letter from licensee forwarding affidavit certifying distribution of Amendment 15 to FSAR. |
| May 22, 1986 | Letter to licensee confirming June 5, 1986, meeting in King of Prussia, PA, to discuss operating experience and lessons learned since receipt of low-power operating license. |
| May 23, 1986 | Letter from licensee forwarding additional information on testability of Bailey 862 solid-state logic modules, in response to April 8, 1986, commitment. |
| May 27, 1986 | Letter from licensee forwarding plant-specific responses to NRC Office of Inspection and Enforcement (IE) Bulletin 85-03, "Motor-Operated Valve Common Mode Failures During Plant Transients Due to Improper Switch Settings," for Salem Units 1 and 2 and Hope Creek, for review. |
| May 27, 1986 | Letter from licensee forwarding Addendum 3 to Revision 4 to "Environmental Qualification Summary Report," reflecting replacement of 53 Tobar Model 32 Series 2 transmitters with Rosemount Model 1153B transmitters, in support of initial criticality and fulfillment of Supplement No. 5. |
| May 29, 1986 | Letter from licensee informing staff that preoperational tests for diesel generator D, in support of OL Condition 2.D(a) and continuation of low-power testing activities, were successfully completed on May 16, 1986. |
| May 30, 1986 | Letter from licensee forwarding application for amendment to License NPF-50, supporting operation with one recirculation loop out of service. General Electric report, "Hope Creek Single Loop Operation Analysis," justifies proposed change and will be incorporated as Appendix 15C to FSAR. |
| June 2, 1986 | Letter from licensee notifying staff that preoperational tests for traversing in-core probe system were successfully completed on May 23, 1986. |



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| June 2, 1986 | Letter from licensee forwarding Addendum 2 to Revision 4 to "Environmental Qualification Summary Report," reflecting reevaluation of electric equipment under Purchase Order M-48, per Supplement No. 5. |
| June 2, 1986 | Letter to licensee forwarding Safety Inspection Report 50-354/86-22. |
| June 3, 1986 | Letter from licensee advising staff of contract with Chem-Nuclear Systems, Inc., to provide temporary radwaste processing services instead of Westinghouse-Hittman Nuclear Services. |
| June 4, 1986 | Letter from licensee requesting approval to operate drywell and suppression chamber purge system per enclosed proposed revision to Technical Specifications. |
| June 5, 1986 | Letter from licensee forwarding LERs 86-014-00 and 86-015-00. |
| June 6, 1986 | Letter to licensee forwarding Safety Inspection Report 50-354/86-24. |
| June 6, 1986 | Letter to licensee forwarding Safety Inspection Report 50-354/86-23. |
| June 9, 1986 | Letter from licensee forwarding LERs 86-016-00 and 86-017-00. |
| June 9, 1986 | Letter to licensee requesting that enclosed table detailing status of NRC review of responses to Generic Letter 83-28, "Required Actions Based on Generic Implications of Salem ATWS Events," be completed with dates for implementation of required actions. |
| June 13, 1986 | Letter from licensee forwarding application for amendment to License NPF-50, revising Technical Specifications because of pending issuance of full-power license. |
| June 13, 1986 | Letter from licensee regarding Bailey 862 solid-state logic module reliability program. |
| June 18, 1986 | Letter to licensee forwarding status of Generic Letter 83-28 review items. |
| June 23, 1986 | Letter to licensee discussing proposed revision to the license condition, "Partial Feedwater Heating." |
| June 24, 1986 | Letter from licensee summarizing the June 23, 1986, teleconference with the staff regarding the solid-state logic module reliability test program. |



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| July 7, 1986 | Letter from licensee requesting revision to the Hope Creek Technical Specifications in response to the staff's June 23, 1986, letter. |
| July 7, 1986 | Letter from licensee submitting the Hope Creek turbine maintenance program based on the turbine manufacturer's probabilities of missile generation. |
| July 11, 1986 | Letter from licensee forwarding a copy of the General Electric proprietary report, "Probability of Missile Generation in General Electric Nuclear Turbines." The licensee requested that this report be withheld from public disclosure. |
| July 24, 1986 | Letter from licensee regarding certain revisions to the Hope Creek Technical Specifications. |



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APPENDIX B

BIBLIOGRAPHY

U.S. Nuclear Regulatory Commission, Generic Letter 83-28, "Required Actions Based on Generic Implications of Salem ATWS Events," July 8, 1983.

---, Generic Letter 86-10, "Implementation of Fire Protection Requirements," April 24, 1986.

---, NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping," August 1981.

---, NUREG-1000, "Generic Implications of ATWS Events at the Salem Nuclear Power Plant," April 1983.

---, Office of Inspection and Enforcement, IE Bulletin 79-27, "Loss of Non-Class 1E Instrumentation and Control Power System Bus During Operation," November 30, 1979.

---, IE Inspection Report 50-354/86-29, June 25, 1986.



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# APPENDIX C

## UNRESOLVED SAFETY ISSUES

In Appendix C to the SER, the staff presented its review of unresolved safety issues (USIs) and their application to Hope Creek. Since that time, USI A-43 has been resolved. The staff's discussion of this USI is given below.

### A-43 Containment Emergency Sump Reliability

This concern is related to the potential for degraded emergency core cooling system performance as a result of thermal insulation debris that may be blown into the suppression pool during a loss-of-coolant accident and cause blockage of the pump suction lines. In the SER, the staff concluded that Hope Creek could be operated before ultimate resolution of this generic issue without endangering the health and safety of the public.

Since the issuance of the SER, the staff has completed its analysis of this issue, which is contained in NUREG-0869, Revision 1, "USI A-43 Regulatory Analysis." The staff has concluded, as a result of this analysis, that no new requirements need to be imposed on licensees and construction permit holders (see Generic Letter 85-22, "Potential for Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage," December 3, 1985). Therefore, this issue is considered resolved for Hope Creek.



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# APPENDIX D

## ACRONYMS AND INITIALISMS

| | |
|---|---|
| ATWS | anticipated transient(s) without scram |
| BWR | boiling-water reactor |
| CFR | Code of Federal Regulations |
| CREFS | control room emergency filtration system |
| EHC | electrohydraulic control |
| FATT | fracture appearance transition temperature |
| FPP | Fire Protection Program |
| FRVS | filtration, recirculation, and ventilation system |
| FSAR | Final Safety Analysis Report |
| GDC | general design criterion(a) |
| GE | General Electric Company |
| HEPA | high efficiency particulate air |
| IE | Office of Inspection and Enforcement |
| LER | Licensee Event Report |
| LOCA | loss-of-coolant accident |
| MCPR | minimum critical power ratio |
| MHC | mechanical hydraulic control |
| NRC | U.S. Nuclear Regulatory Commission |
| OL | operating license |
| PAP | power ascension program |
| PSE&G | Public Service Electric and Gas Company |
| RFT | reduced feedwater temperature |
| RG | regulatory guide |
| SER | Safety Evaluation Report |
| SSLM | solid-state logic module |
| TMI | Three Mile Island |
| USI | unresolved safety issue |
| UT | ultrasonic testing |

APPENDIX E

PRINCIPAL STAFF CONTRIBUTORS

This supplement to the Safety Evaluation Report is a product of the NRC staff and its consultants. The NRC staff members listed below were principal contributors to this report.

| Name | Title | Branch |
|---|---|---|
| Richard Becker | Reactor Engineer | Facility Operations |
| Jay Lee | Health Physicist | Plant Systems |
| Felix Litton | Mechanical Engineer | Engineering |
| Barry Marcus | Electrical Engineer | Electrical, Instrumentation and Control Systems |
| Carl Schulten | Reactor Engineer | Facility Operations |
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| John Tsao | Mechanical Engineer | Engineering |
| Frank Witt | Mechanical Engineer | Plant Systems |



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# APPENDIX S

# REVIEW OF POWER ASCENSION PROGRAM ACCELERATION



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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

January 22, 1986

Docket No. 50-354

Mr. Corbin A. McNeill, Jr., Vice President-Nuclear
Public Service Electric & Gas Company
Nuclear Administration Building
P. O. Box 236
Hancocks Bridge, New Jersey 08038

Dear Mr. McNeill:

SUBJECT: HOPE CREEK POWER ASCENSION PROGRAM TEST MODIFICATIONS

By letters dated August 21, September 20,23 and 30, 1985, PSE&G submitted for staff review a number of proposed test modifications to the Hope Creek Power Ascension Program. The proposed test modifications were submitted for staff review as part of a program to accelerate power ascension testing for Hope Creek. Additionally, the Hope Creek accelerated power ascension program is the lead plant in an effort by General Electric to similarly accelerate the power ascension program on a number of other boiling water reactors.

The staff has completed its review of the following test modifications proposed in the above referenced letters:

1. Automatic Load Following
2. Test No. 17 - Core Performance
3. Test No. 27 - Recirculation Flow Control System
4. Test No. 12 - RCIC System
5. Test No. 13 - HPCI System
6. Test No. 14A - Selected Process Temperatures
7. Test Condition 4 - Natural Circulation Operation
8. Test No. 20 - Pressure Regulator
9. Test No. 21A - Feedwater System Response Testing

The safety evaluation detailing each review is attached. For those test modifications which the staff has accepted (Items 1,4,5,6,8 in part and 9 in part), we request that the Hope Creek FSAR be amended.

Sincerely,

Robert Bernero, Director
Division of BWR Licensing
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SAFETY EVALUATION

HOPE CREEK GENERATING STATION

POWER ASCENSION TEST PROGRAM ACCELERATION

This safety evaluation describes the staff's review of a number of Hope Creek Generating Station - Power Ascension Program (PAP) test modifications proposed by Public Service Electric and Gas Company (PSE&G). These proposed test modifications were submitted for staff review as part of a program to accelerate power ascension testing for Hope Creek. The Hope Creek accelerated power ascension testing program is also the lead plant for generic changes for acceleration of the traditional General Electric Boiling Water Reactor power ascension test program.

The proposed test modifications discussed in this safety evaluation were submitted by letters dated August 21, September 20, 23, and 30, 1985, (R. Mittl, PSE&G to W. Butler, NRC). This safety evaluation discusses the following tests:

1. Automatic Load Following
2. Test No. 17 - Core Performance
3. Test No. 27 - Recirculation Flow Control System
4. Test No. 12 - RCIC System
5. Test No. 13 - HPCI System
6. Test No. 14A - Selected Process Temperatures
7. Test Condition 4 - Natural Circulation Operation
8. Test No. 20 - Pressure Regulator
9. Test No. 21A - Feedwater System Response Testing

Discussion of the above tests follows:

1. DELETION OF AUTOMATIC LOAD FOLLOWING (ALF) TESTING

The applicant proposed deleting all testing of the ALF function from the power ascension test program. The ALF portion of the Recirculation Flow Control System is a non-safety related function which is not intended to be used at present. Because the Automatic function is not going to be used, it is unnecessary to test this function of the Recirculation Flow Control System. If at some future time the ALF function is to be used, it can be tested at that time. The Automatic mode of operation should be physically disconnected from the Master Flow Controller and wired so that the system is in manual control in both Master Flow Controller Positions to prevent inadvertent use of the Automatic function. A similar deletion has been granted at other plants (e.g., Grand Gulf). Therefore, if the Automatic function of the Master Flow Controller is physically disabled, deletion of testing of the ALF function is acceptable.


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2. SUBSTITUTION OF SURVEILLANCE TEST FOR CORE PERFORMANCE STARTUP TEST (TEST NO. 17)

The applicant proposed substitution of the Core Performance Technical Specification Surveillance test for the Core Performance Startup test to reduce duplication of testing.

The applicant maintains that adherence to plant technical specifications meets the objectives of startup test No. 17 and the intent of Regulatory Guide 1.68 (Revision 2, August 1978), Appendix A, paragraph 5.b. The staff believes that the performance of the fuel thermal margins technical specification surveillances simply allows operational personnel to determine that the plant is operating within its required limiting condition for operation and does not constitute a determination "that steady-state core performance is in accordance with design" (Reg. Guide 1.68). The objective of Startup Test No. 17 and Regulatory Guide 1.68 is that an engineering evaluation be performed to evaluate the actual thermal and hydraulic parameters against design expectations for these parameters and that it, in turn, receives the appropriate level of management attention. Accordingly, the applicant's proposed substitution of the Core Performance Technical Specification surveillance test for the Core Performance Startup test is not acceptable.

An alternative to the applicant's proposed substitution exists. It is the intent of Regulatory Guide 1.68, Revision 2, to utilize plant procedures wherever possible, both to validate the procedures and provide training for plant personnel in their use. In Section C, Regulatory Position, Part 2, of Regulatory Guide 1.68, it states, "The overall test program should also include surveillance tests necessary to demonstrate the proper operation of interlocks, setpoints, and other protective features, systems, and equipment required by the technical specifications." On the basis of the regulatory guidance in place, the staff feels that both the duplication of tests can be avoided and the more technically extensive review and approval required for startup tests can be accomplished by writing the detailed core performance startup test procedure such that the surveillance procedure is incorporated to direct data taking and recording. By this method, both regulatory intent to utilize plant procedures and the administrative review, evaluation, and approval controls for startup tests will be included. Because this regulatory position already exists, staff approval of this proposed change would not be necessary.



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3. SIMPLIFICATION OF THE RECIRCULATION FLOW CONTROL SYSTEM (TEST NO. 27)

The applicant proposed deleting testing of the ALF portion of this system and utilizing experience from other plant startups to modify the calibration and testing of the recirculation flow control system to reduce the number of test points used to optimize control setting and check stable operation.

The staff has discussed and accepted (in Item 1) the deletion of ALF testing in the Recirculation Flow Control System. The selection of the number of test points and their distribution in testing space is a level of detail implementation which does not appear in the FSAR test abstract, but may be reviewed at the Regional level in the detailed test procedure. Because the requested utilization of experience for bench calibration of controllers to reduce the number of system test points does not change the test objectives, summary test method or acceptance criteria of the test abstract, the staff finds the proposed changes conceptually acceptable.

As long as the applicant can demonstrate compliance with the acceptance criteria throughout the appropriate system control range, test simplification is encouraged.

4. & 5. RCIC AND HPCI SYSTEM TESTS (TESTS 12 AND 13)

The applicant proposed deleting tuning of the HPCI and RCIC controllers during the low power testing. Currently, tuning of the controllers for the RCIC and HPCI is performed at both low pressure and near rated reactor pressure. The applicant indicates that testing experience with recently commercialized BWRs has demonstrated that a best estimate controller setting may be selected from previous test data and used to bench calibrate the controllers during preoperational testing. This preliminary bench controller tuning is sufficient for low pressure testing and final tuning near rated reactor pressure. Experience at previous plants has shown that controller tuning at low pressure condition does not result in optimum performance at higher pressures. The applicant is requesting deletion of controller tuning at the low reactor pressure condition, relying on controller tuning or optimization at or near rated reactor pressure. Testing at low reactor pressure with the rated reactor pressure controller setting will be completed to confirm acceptable, but, perhaps, less than optimized system performance at low pressure conditions.

The staff has reviewed the applicant's requested change to the method of testing these two systems. The staff recognizes that it would be unlikely that operation of these two systems could be optimized over the entire pressure range for which they are expected to operate. Selecting rated pressure conditions for the point of optimization is reasonable and acceptable to the staff because acceptable operation at low reactor pressure will be confirmed with the rated pressure controller setting.

6. SELECTED PROCESS TEMPERATURES (TEST NO. 14A)

The applicant is requesting the substitution of the Technical Specification surveillance test for startup of an idle reactor recirculation loop for portions of the selected process temperatures startup procedure and deletion of determination of the low pump speed limit by utilizing a fixed value established from results of this test on previous plants.

The staff has reviewed the applicant's proposed changes and notes that the startup test monitors process temperatures to assure the technical specifications are not exceeded. The staff believes that the technical specification surveillances accomplish the same purpose. Additionally, the staff notes that testing performed at Limerick and Susquehanna has demonstrated ample margins for stratification.

Accordingly, the staff finds this requested substitution acceptable. The Region staff will confirm during future test procedure review that the surveillance test has been properly incorporated into the procedure and that all necessary test data will be obtained.

The staff found that the request to delete the determination of the low pump speed limit in favor of using a fixed limit derived from previous plant tests was not supported by sufficient information. The applicant supplied additional information on the startup testing for the Brunswick Steam Electric Plant, Unit 1 and Edwin I. Hatch, Unit 2 which was reviewed by the staff. This information was contained in the Final Summary Report - Edwin I. Hatch Unit 2 - Startup Test Results, NEDO-24734, R. W. Turkowski and W. Yee, October 1979, and Brunswick Unit 1 - Startup Test Results - Final Summary Report, NEDO-24562, J. B. Poppel, November 1977. This additional information and regulation-mandated startup reports of other plants were reviewed and allowed the staff to confirm that a fixed low pump speed limit could be selected (based on previous plant testing) which assures adequate coolant mixing and, therefore, acceptable temperature differentials in the lower plenum.

7. NATURAL CIRCULATION OPERATION - TEST CONDITION NO. 4

The applicant requested deletion of all testing at Test Condition 4 (approximately 50% power and 30% core flow) based on the stated premise that the applicant does not intend to operate in this domain of the power/core flow map and that the only way this domain would be approached would be an abnormal operational transient (trip of two recirculation pumps) and resulting natural circulation. The plant technical specifications require that with no reactor coolant system recirculation loops in operation, the operator must take immediate action to reduce thermal power to less than or equal to that allowed by the Core Thermal Power to Core Flow Map within 2 hours and, further, the operator must take action to place the plant in at least startup within 6 hours. Therefore, the applicant states that testing need not be done at Test Condition 4.

The staff felt that the applicant's justification for deleting testing at Test Condition 4 was inadequate. The reactor manufacturer, GE, submitted additional information to supplement the applicant's justification by letter from J. F. Klapproth, GE, to R. A. Becker, NRC, dated November 6, 1985. The staff included this information in addition to the applicant's submission in its review.

The staff finds that the justification for deleting testing at Test Condition 4 is insufficient. The staff considers that the technical specifications define the "possible operating modes," whether or not the applicant foresees their utilization at the commencement of the plant's 40-year life. The length of time allowed at or near natural circulation is long compared to a phenomenon which could generate a safety concern such as stability. The staff agrees with the applicant that the generic natural circulation and stability have been sufficiently confirmed in recent domestic and foreign commercialized BWRs. However, the staff believes the testing encompassed by Regulatory Guide 1.68, Appendix A, Section 5, third introductory paragraph refers to plant specific confirmatory testing and Test Condition 4 represents one of the "extremes of possible operating modes" referred to in the Regulatory Guide. Therefore, thorough testing dictates equipment specific testing at the natural circulation, Test Condition 4 and the staff finds the elimination of testing at Test Condition 4 to be unacceptable.

8. PRESSURE REGULATOR (TEST NO. 20)

The applicant proposed to modify the pressure regulator test to: (a) delete performing this test at Test Condition 4 (TC4), (b) delete the backup pressure regulator takeover testing at TC5, and (c) delete ALF mode tests at TC3 and TC6.

Regarding item (a) above, the staff evaluation of Natural Circulation Operation - Test Condition No. 4 (Item No. 7 in this safety evaluation) concluded that elimination of testing at TC4 is unacceptable. Accordingly, the proposal to delete testing of the pressure regulator at TC4 is not acceptable.

Regarding item (b) above, the applicant requested the deletion of backup pressure regulator takeover testing at TC5. The system design includes redundant pressure regulators set at slightly different operating pressures. If the pressure regulator operating at the lower pressure were to fail, the higher pressure regulator would assume pressure control. This design function is verified over the spectrum of operating conditions. Currently, this is performed at TC1 through TC6 which cover the power range from low to rated steam flow.

The staff believes that the only variable of concern is total steam flow or, indirectly, plant power level because the pressure regulator maintains fixed pressure for various total flows. For a fixed power level, variation



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of core flow would have no impact on the pressure regulator. Examination of the operational power/flow map indicates that TC3 covers a similar but larger power span than TC5 and, therefore, provides a redundant test point for testing of the backup pressure regulator takeover. Because of the redundant nature of this portion of the pressure regulator test at TC5 and TC3, the staff believes that testing of the takeover of the backup pressure regulator at TC5 may be deleted without affecting the objectives of the test or loss of confirmation of system performance over the entire operating range.

Regarding item (c) above, the applicant requested deleting all testing associated with the ALF mode of plant operation. This request was addressed and accepted in our review contained in Automatic Load Following (Item No. 1 of this safety evaluation).

9. FEEDWATER SYSTEM RESPONSE TESTING (TEST NO. 21A)

The applicant proposed to modify the Feedwater Control System Test by: (a) deleting testing at TC4 and (b) deleting ALF testing.

Regarding item (a), the staff evaluation of Natural Circulation Operation - Test Condition 4 (Item No. 7 in this safety evaluation) concluded that elimination of testing at TC4 is unacceptable. Accordingly, the proposal to delete feedwater system response testing at TC 4 is not acceptable.

Regarding item (b), the applicant has requested deleting all testing associated with the ALF mode of plant operation. This request was addressed and accepted in our review contained in Automatic Load Following (Item No. 1 of this safety evaluation).



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Docket No. 50-354

Mr. Corbin A. McNeill, Jr., Vice President-Nuclear
Public Service Electric & Gas Company
Nuclear Administration Building
P. O. Box 236
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Dear Mr. McNeill:

Subject: Hope Creek Power Ascension Program Test Modifications

By letters dated August 21, October 4 and 17, November 6, and December 9, 1985, PSE&G submitted, for staff review, a number of proposed test modifications to the Hope Creek Power Ascension Program. The proposed test modifications are part of a program to accelerate power ascension testing for Hope Creek. Additionally, Hope Creek is the lead plant in an effort by General Electric to promote accelerated power ascension programs on a number of other boiling water reactors.

The staff has completed its review of the following test modifications proposed in the above referenced letters:

1. Test No. 24 - Relief Valves
2. Test No. 28E - Recirculation System Cavitation
3. Test No. 31 - Drywell Piping Vibration
4. Test No. 25 - Turbine Trip and Generator Load Rejection
5. Test No. 28D - Recirculation Pump Runback Test
6. Test No. 3 - Elimination of Fuel Chambers During Fuel Loading
7. Test No. 11 - Process Computer Test Simplification
8. Test No. 16 - TIP Uncertainty
9. Test No. 1 - Chemical Radiochemical Test Simplification
10. Test No. 32 - Reactor Water Cleanup System
11. Test No. 28B - Two Pump Recirculation Pump Trip Test

The safety evaluation detailing each review is attached. For those test modifications which the staff has accepted (Items 1, 2, 4, 5, 6, 7, 8, 9 and 10 in part), we request that the Hope Creek FSAR be amended to reflect these changes.

Sincerely,

Robert Bernero, Director
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SAFETY EVALUATION

HOPE CREEK GENERATING STATION

POWER ASCENSION TEST PROGRAM ACCELERATION

This safety evaluation describes the staff's review of a number of Hope Creek Generating Station - Power Ascension Program (PAP) test modifications proposed by Public Service Electric and Gas Company (PSE&G). These proposed test modifications were submitted for staff review as part of a program to accelerate power ascension testing for Hope Creek. Hope Creek is also the lead plant for generic changes for acceleration of the traditional General Electric Boiling Water Reactor power ascension test program.

The proposed test modifications discussed in this safety evaluation were submitted by letters dated August 21, October 4 and 17, November 6 and December 9, 1985. This safety evaluation discusses the following tests:

1. Test No. 24 - Relief Valves
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7. Test No. 11 - Process Computer Test Simplification
8. Test No. 16 - TIP Uncertainty
9. Test No. 1 - Chemical Radiochemical Test Simplification
10. Test No. 32 - Reactor Water Cleanup System
11. Test No. 28B - Two Pump Recirculation Pump Trip Test

Discussion of the above tests follows:

1. TEST NO. 24 - RELIEF VALVES

The purpose of Test number 24 - Relief Valves, is to demonstrate the operability of the relief valves, i.e., that they can be opened and closed manually, that they reseat properly, and that there are no blockages in the relief valve piping. This testing is currently planned to be performed at low (250-500 psig) pressure during heatup and at rated reactor pressure between Test Conditions 2 and 3. By letter dated October 17, 1985, the applicant proposed to delete the operability test at low pressure and between Test Conditions 2 and 3. Alternatively, the applicant proposed verifying operability between 10% and 20% power at Test Condition 1.

The Hope Creek Safety Relief Valves (SRVs) are manufactured by Target Rock Corporation. Actuation of the relief valves at low pressure has been identified as a contributor to valve seat damage caused by the valves' reseating against abnormally low pressure. Therefore, operation of the valves at low pressure should be avoided whenever possible. Furthermore, from an operational point of view, conducting the test at a steam flow greater than the capacity of a



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relief valve (typically 5-7%) will significantly enhance plant pressure control during the transient. Finally, protection from the effects of over-pressure transients which may occur prior to relief valve operability testing will be assured through compliance with Technical Specification 4.5.1.d which requires that all automatic depressurization system (ADS) valves be manually opened within twelve hours of reaching a steam dome pressure of 100 psig. The ADS valves by themselves provide sufficient relief capacity to mitigate the relatively mild overpressure transients which could occur at less than 20% power. For the reasons stated above, the proposed change to test the SRVs at Test Condition 1 is acceptable.

2. TEST NO. 28E - RECIRCULATION SYSTEM CAVITATION

Test Number 28E, Recirculation System Cavitation, verifies that no recirculation system cavitation will occur in the operable region of the power-flow map. Currently, this test is planned to be performed at Test Condition 3 by lowering power at high recirculation flow until a recirculation pump runback occurs, or the plant is at approximately 18% power and no runback has occurred, or there is indication of cavitation. By letter dated October 17, 1985, the applicant proposed that the testing be simplified by temporarily bypassing the cavitation interlock to prevent runback of the recirculation pumps.

Acceptable response of the system near the cavitation region is determined by analyzing test data and comparing to acceptance criteria which define the required system performance. For the recirculation system cavitation test, the recirculation runback logic is required to be demonstrated to have settings adequate to prevent operation in areas of potential cavitation. This may be demonstrated without requiring that recirculation runback occur by temporarily bypassing the runback logic during the test. Power can then be reduced by inserting control rods when cavitation is indicated. The cavitation interlock logic can be monitored to ensure that it is actuated at the correct setpoint.

With appropriate placement of the jumper on the cavitation interlock, no other recirculation pump runback logic feature will be affected. In addition, should a feedwater transient occur during the performance of this test, the operators can manually run back recirculation pump speed as necessary to prevent cavitation.

Cavitation interlock setpoints are designed to allow maximum operation in the power-flow map but are conservatively set to assure that no recirculation system cavitation occurs. The proposed simplified testing will verify that cavitation does not occur at or above the cavitation interlock setpoint while eliminating unnecessary core flow reductions during the test. Similar testing simplification was used at Limerick and Susquehanna. The proposed change to the Recirculation System Cavitation test is acceptable.

3. TEST NO. 31 - DRYWELL PIPING VIBRATION

Test Number 31 was intended to evaluate the dynamic response of the main steam and recirculation piping to various transients in terms of vibrational characteristics and dynamic deflection. By letter dated October 17, 1985, the applicant proposed eliminating two of these transients from this test series. These transients are the recirculation pump trips and restarts. The applicant's justification for eliminating these tests is that previous startup results on similar plants indicate that vibration and deflection measurements of recirculation piping during recirculation pump trips and restarts are always well below the prescribed limits. As part of this justification, the applicant submitted a quantitative comparison of recirculation piping system parameters on Hope Creek and Susquehanna Unit 1. Table 2 in the applicant's October 17, 1985, letter shows that these parameters are almost identical. In addition, Table 1 of the applicant's letter shows that measured vibrations for the recirculation pump trip/restart transients on Susquehanna Unit 1 were well below the allowable.

The basic objective of all piping preoperational testing is to confirm the as-built condition of piping systems in each plant. Although the design of such systems may be identical from plant to plant, fabrication, installation, inspection and quality assurance of the piping and the pump could introduce differences or defects which may affect the dynamic characteristics of the piping systems. Therefore, the staff's position is that all of the preoperational piping tests which the applicant, in the FSAR, agreed to conduct shall be completed and that the proposed modifications to these tests are not acceptable.

4. TEST NO. 25 - TURBINE TRIP AND GENERATOR LOAD REJECTION

By letter dated August 21, 1985, the applicant proposed a) changing the generator load rejection test at Test Condition 2 to a turbine trip test and b) deleting the turbine trip test at Test Condition 3.

The staff finds these proposed changes acceptable. Regarding changing the generator load rejection test at Test Condition 2 to a turbine trip test at Test Condition 2, the staff notes that the purpose of this test segment, namely to test turbine bypass system performance, can be accomplished by either a turbine trip test or a generator load rejection test. Accordingly, this modification is acceptable. Regarding deleting the turbine trip test at Test Condition 3, the applicant indicates that performing the already planned turbine generator full load rejection test at 100% power required by Regulatory Guide 1.68 is more bounding with respect to severity than the turbine trip test at Test Condition 3. The applicant provided comparative date for these two tests for five boiling water reactors to support that position.

The staff has reviewed the evaluation and supporting data and agrees with the applicant that the regulatory guidance outlined in Regulatory Guide 1.68, Revision 2, related to plant testing involving the turbine generator is satisfied by performing a full load rejection/turbine trip test at Test Condition 6 (100% power). Therefore, the staff finds it acceptable to delete the turbine trip test at Test Condition 3.

5. TEST NO. 28D - RECIRCULATION PUMP RUNBACK TEST

By letter dated October 4, 1985, the applicant requested that the recirculation pump runback test, currently planned as a separate test at Test Condition 3, be combined with the feedwater pump trip test at Test Condition 6.

The staff finds this change acceptable for the following reasons. The recirculation pump runback feature is included in the design to avoid unnecessary scrams due to low water level when a feedwater pump inadvertently trips. When performed at Test Condition 3, the runback is initiated by a simulated feedwater pump trip signal. If performed at Test Condition 6 as part of the feedwater pump trip test, demonstration of the runback circuit would be obtained as an actual integrated test. Therefore, some testing advantage is obtained without compromising any safety objectives.

6. TEST NO. 3 - ELIMINATION OF FUEL LOADING CHAMBERS DURING FUEL LOADING

By letter dated December 9, 1985, the applicant proposed simplifying the fuel loading procedure by replacing the fuel loading chambers (FLCs) with source range monitor (SRM) instrumentation. Additionally, the startup sources will be positioned in their alternate locations (to be closer to the SRMs) and the fuel loading sequence will be modified such that initial fuel loading will begin between an SRM detector and a neutron source. Fuel will be loaded in a spiral pattern around the SRM until the core is fully loaded.

Test No. 3 is conducted in the fuel loading phase of initial operations. It has usually been necessary in past fuel loadings to use FLCs in addition to SRM detectors to achieve Technical Specification required count rates with fuel in the core. A number of utilities have, in the past, requested and been granted reload fuel loading operations in which a small number of fuel assemblies are loaded before the usual required count rate on SRMs or FLCs are achieved. This initial loading is sufficient to provide the needed count rate. Additionally, the permitted (small) number of assemblies can not become critical, even with all control rods removed.

The proposed modifications to Hope Creek Test No. 3 are to permit the loading of 16 assemblies without (necessarily) meeting the usual required 0.7 counts/sec for the SRMs or FLCs. This is based on analysis (by General Electric) which shows that this array would not be critical (with rods out) and would provide the necessary count rate. This change would make it unnecessary to use FLCs, which interfere with operations, and would permit the use of the standard SRMs alone. The procedure places the sources in alternate locations, close to an SRM and uses a spiral loading pattern around the initial source-SRM and 16 assembly locations rather than around the core center.

These procedures are compatible with a number of previously approved reload procedures. The criticality calculations are done with standard methodology. They are consistent with other analyses reviewed in this area. Tests on other reactor startups indicate that required count rates should be achieved. The Technical Specifications will require (after loading 16 assemblies) the usual count rate on at least one SRM (the SRM near the initial loading). The other SRMs will be checked with a source, as has been approved for other reactors, until they reach a suitable count rate.



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The staff notes that the Technical Specifications should be revised to specify that one of the operable SRMs must be in an area that has fuel loaded around it on one side. Specifically, we require that two SRMs be operable and continuously indicating in the control room. One of the SRMs will be in the quadrant where fuel is being loaded and the other will be in an adjacent quadrant. One of these two SRMs will be in an area in which fuel has been loaded. The minimum count rate must be met for at least one of the SRMs. With this additional requirement, the applicant's proposal is acceptable.

7. TEST NO. 11 - PROCESS COMPUTER TEST SIMPLIFICATION

By letter dated November 6, 1985, the applicant proposed simplifying Test No. 11. Test No. 11 involves the testing of the Process Computer and its programs. OD-11 is one of these programs and deals with the area of fuel pellet-clad interaction monitoring (Preconditioning Interim Operating Management Recommendations, (PCIOMR)). The program assists in implementing PCIOMR to prevent this type of fuel failure mechanism during operation. However, with barrier fuel (as used at Hope Creek), General Electric Company (GE) (the fuel facricator) has removed the PCIOMR procedures from the operation plans since they are no longer needed. Accordingly, the applicant has proposed the removal of the OD-11 test from the startup program. Our review has indicated that there is no need for PCIOMR monitoring for this fuel. The removal of OD-11 monitoring is acceptable.

8. TEST NO. 16 - TIP UNCERTAINTY

By letter dated November 6, 1985, the applicant proposed deleting Test No. 16 from the power ascension program. Test No. 16 measures the Traversing Incore Probe (TIP) uncertainty. The uncertainty is composed of geometry effects and random noise. These are determined by comparing symmetric pairs of TIP readings and repeated traverses of common TIP tubes. The criterion for first cycle TIP uncertainty tests is that uncertainty should be less than 6 percent. This is the value which, if used in the uncertainty analyses for GETAB (rather than the 2.6 percent value normally used in first cycle), would increase the power density value sufficiently to increase the safety limit minimum critical power ratio (MCPR) by 0.01. Previous tests in other reactors (including LaSalle 1 and Susquehanna 1) have always provided a TIP uncertainty well below 6 percent. Furthermore, the uncertainty is lower when using the recently introduced TIP gamma detector rather than the usual neutron detector since the gamma system is less sensitive to geometry errors. For these reasons, the applicant has proposed deleting this test.

TIP operability is determined in preoperational testing and during power ascension power distribution measurements and tests of the Process Computer. Previous tests in other reactors have indicated no problem in the TIP uncertainty area, and the gamma detectors to be used have lower uncertainty parameters. The Hope Creek system should be well below the criteria. The safe operation of the plant will not be affected by deleting this test. Accordingly, the proposed deletion of Test No. 16 is acceptable.



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9. TEST NO. 1 - CHEMICAL RADIOCHEMICAL TEST SIMPLIFICATION

By letter dated December 9, 1985, the applicant proposed to substitute plant surveillance procedures for the chemistry and radiochemistry monitoring requirements. Additionally, the applicant proposed deleting the integrated performance testing of the reactor water cleanup system (RWCU) and condensate demineralizer system at Test Condition 3.

The purpose of Test No. 1 is to demonstrate that the plant water chemistry and radiochemistry are within limits during the power ascension test program and also to demonstrate the design capability of the plant chemistry system. It is proposed by the applicant to substitute the plant surveillance procedure CH-TI.ZZ-012(Q), Chemistry Sampling and Surveillance Procedure, for the chemistry and radiochemistry monitoring requirements of Test No. 1. This surveillance procedure, based on BWR Water Chemistry Guidelines (BWR Owners Group/EPRI report dated April 1, 1984), is used to ensure that plant water chemistry meets fuel warranty limits. The improved water chemistry can enhance fuel performance and minimize radiation field buildup on out-of-core surfaces. The surveillance procedure insures that plant water chemistry meets the limits specified by Technical Specification 3.4.4 and the General Electric fuel warranty. The surveillance procedure limits are at least as restrictive as those of Test No. 1. Since the surveillance procedure will be used during normal operation it would be prudent to also use this procedure during power ascension testing. It is therefore acceptable to substitute the plant surveillance procedure CH-TI-ZZ012(Q) for Test No. 1, Chemical and Radiochemical, for monitoring plant water chemistry and radiochemistry during power ascension testing. Although this substitution is acceptable, we encourage the applicant to review the results of the surveillance with the same management attention which would have been given the review of the startup test.

In Test No. 1, the RWCU and condensate demineralizer systems are performance tested to demonstrate that they meet design specifications at Test Conditions 3 (low power) and 6 (rated power/flow). Performance testing the RWCU and condensate demineralizer systems at full power and flow, Test Condition 6, will demonstrate the ability of these systems to adequately control coolant chemistry at the most demanding plant operating condition. Performance testing at a lower power, Test Condition 3, can verify procedures and provide preliminary data, but is not required to demonstrate meeting design specifications. Therefore, it is acceptable to delete the RWCU and condensate demineralizer systems integrated performance testing requirement of Test No. 1 at Test Condition 3 as proposed by the applicant.

10. TEST NO. 32 - REACTOR WATER CLEANUP SYSTEM

By letter dated, October 17, 1985, the applicant proposed modifying Test No. 32 - Reactor Water Cleanup System. The purpose of Test No. 32, is to demonstrate the operability of reactor coolant system purification and cleanup systems during low power testing. It is proposed by the applicant to a) delete the RWCU non-regenerative heat exchanger (NRHX) flow test in the blowdown mode, b) delete the bottom head flow rate calibration from the power ascension test program, c) perform the reactor water cleanup system (RWCU) pump net positive

suction head (NPSH) test under cold conditions during preoperational testing, and d) perform the non-regenerative heat exchanger (NRHX) flow test in the normal mode during Test Condition 1 instead of during Test Condition Heatup. The RWCU system temperature and flow measurements will be obtained during the normal operating mode to demonstrate the heat exchange capability of the NRHX. However, during the blowdown mode, the regenerative heat exchanger capacity is decreased as a result of partially bypassing a portion of the reactor coolant system return flow to the main condenser or radwaste system. This could automatically isolate the RWCU system on NRHX high outlet temperature. RWCU isolation is not desirable during heatup when water chemistry is critical and when excess reactor coolant needs to be discharged. Therefore, the RWCU NRHX flow test in the blowdown mode should remain in Test No. 32.

The applicant proposed that the bottom head flow rate calibration be performed after completion of power ascension testing. This test is not critical to demonstrating the performance of the RWCU system. Therefore, it is acceptable to delete the bottom head flow calibration from Test No. 32 and postpone it until after completion of power ascension testing.

The applicant proposed to determine RWCU pump NPSH during preoperational testing under cold conditions. Calculations can be performed to extrapolate NPSH from preoperational cold conditions to operational conditions to demonstrate compliance with acceptance criteria. Therefore, it is acceptable to determine RWCU pump NPSH during preoperational testing.

The applicant proposed to perform the RWCU flow test for the NRHX at Test Condition 1 instead of at Test Condition Heatup. RWCU operation in the blowdown mode is important during heatup since during this phase, part of the reactor coolant will be bypassed to the main condenser or radwaste system. Therefore, the RWCU flow test for the NRHX should remain in Test Condition Heatup in Test No. 32. The applicant's proposal is not acceptable.

11. TEST NO. 28B - TWO PUMP RECIRCULATION PUMP TRIP TEST

By letter dated October 4, 1985, the applicant requested deleting the two pump trip and flow coastdown at Test Condition 3 and using the data obtained on the integrated systems Generator Load Rejection Test at Test Condition 6 to obtain the two pump trip and flow coastdown data to satisfy Regulatory Guide 1.68, Revision 2. The staff found this change to be unacceptable for the following reason. By letter dated August 21, 1985, the applicant requested to delete the turbine trip test at Test Condition 3 (see Test No. 25 - Turbine Trip and Generator Load Rejection, Item 4 in this safety evaluation). The turbine trip test deletion at Test Condition 3 was found acceptable based, in part, on the implicit condition that safety and accident mitigation features were tested before full power was attained. Explicitly, this conclusion was based, in part, on the fact that the two recirculation pump trip and flow coastdown would be performed at Test Condition 3 prior to the load rejection test at full power. Therefore, the two pump trip and flow coastdown should be performed at Test Condition 3 to test this accident mitigation feature before full power is attained.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MAR 20 1985

Docket No. 50-354

Mr. Corbin A. McNeill, Jr., Vice President-Nuclear
Public Service Electric & Gas Company
Nuclear Administration Building
P. O. Box 236
Hancocks Bridge, New Jersey 08038

Dear Mr. McNeill:

Subject: Hope Creek Power Ascension Program Test Modifications

By letters dated August 21, September 20, October 17, and December 9, 1985, PSE&G submitted, for staff review, a number of proposed test modifications to the Hope Creek Power Ascension Program. The proposed test modifications are part of a program to accelerate power ascension testing for Hope Creek. Additionally, Hope Creek is the lead plant in an effort by General Electric to promote accelerated power ascension programs on a number of other boiling water reactors.

The staff has completed its review of the following test modifications proposed in the above referenced letters:

1. Test No. 19 - Core Power/Void Mode Response
2. Test No. 9 - LPRM Calibration
3. Test No. 10 - APRM Calibration
4. Test No. 23A - MSIV Functional Test
5. Test No. 5 - Control Rod Scram Time Testing at Hot Standby with Full Reactor Scram Control Rod Time Testing
6. Test No. 23B - MSIV Full Isolation

The safety evaluation detailing each review is attached. For those test modifications which the staff has accepted (Items 1, 3 in part, and 4), we request that the Hope Creek FSAR be amended to reflect those changes. This concludes our review of the submitted Hope Creek Power Ascension Program Test Modifications.

Sincerely,

Robert Bernero, Director
Division of BWR Licensing
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc w/enclosure:
See next page



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## SAFETY EVALUATION

## HOPE CREEK GENERATING STATION

## POWER ASCENSION TEST PROGRAM ACCELERATION

This safety evaluation describes the staff's review of a number of Hope Creek Generating Station - Power Ascension Program (PAP) test modifications proposed by Public Service Electric and Gas Company (PSE&G). These proposed test modifications were submitted for staff review as part of a program to accelerate power ascension testing for Hope Creek. Hope Creek is also the lead plant for generic changes for acceleration of the traditional General Electric Boiling Water Reactor power ascension test program.

The proposed test modifications discussed in this safety evaluation were submitted by letters dated August 21, September 20, October 17, and December 9, 1985. This safety evaluation discusses the following tests:

1. Test No. 19 - Core Power-Void Mode Response Calibration
2. Test No. 9 - Local Power Range Monitor
3. Test No. 10 - Average Power Range Monitor Calibration
4. Test No. 23A - MSIV Functional Tests
5. Test No. 5 - Control Rod Scram Time Testing at Hot Standby with Full Reactor Scram Control Rod Time Testing
6. Test No. 23B - MSIV Full Isolation

Discussion of the above tests follows:

1. TEST NO. 19 - CORE POWER-VOID MODE RESPONSE

In a letter dated August 21, 1985, the applicant proposed deletion of the Core Power-Void Mode Response test. This is a test where the core power response to both the insertion and removal of a high reactivity worth control rod and small step changes in reactor pressure is obtained and examined for stable behavior. The applicant states that this test may be eliminated based largely on experience gained during other plant startups, and that the Pressure Regulator Test (Test No. 20) performed at identical test conditions provides better information to determine dynamic stability in large boiling water reactors (BWRs). The applicant submitted the staff's acceptance and safety evaluation report (SER) for the Licensing Topical Report, "Thermal Hydraulic Stability Amendment to GESTAR II," Rev. 6, Amendment 8, NEDE-24011 as supporting technical information.

The significant issue with this test is whether it produces significant testing not duplicated by the Pressure Regulator Test. Results of these two tests in three recently declared commercial BWRs were examined. It was possible to confirm from examination of the testing matrix and the test descriptions that identical pressure response testing was accomplished


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in both core power-void mode response and pressure regulator tests. However, beyond confirmation of duplication of testing, the test reports were of little value because they only included discussion of meeting the acceptance criteria for the respective tests and did not provide comparative data. One plant which did include some reduced data indicated very little response to control rod movement but significant response to pressure changes. The results of both tests met the acceptance criteria and confirmed stability.

The supporting technical material related to Topical Report NEDE-24011 was reviewed. From the review of this supporting material, the staff finds that (a) a considerable body of BWR special stability testing exists in addition to the successful stability testing of the newly commercial BWR plants and (b) small perturbations of reactor pressure have been demonstrated to be a simple testing approach to determine reactor stability (see L. A. Carmichael and R. O. Niemi, "Transient and Stability Tests at Peach Bottom Atomic Power Station, Unit 2, End of Cycle 2," Electric Power Research Institute, 1978 (EPRI Np. 564)). Therefore, the staff concludes that control rod oscillation testing to establish reactor stability is unnecessary because it is duplicative of testing performed at similar plants and produces lesser quality stability confirmation than reactor pressure perturbation techniques. Further, the staff concludes that it is unnecessary to duplicate the pressure cycling tests of the Pressure Regulator Test and it is, therefore, acceptable to delete the Core Power-Void Mode Response Test.

2. TEST NO. 9 - LOCAL POWER RANGE MONITOR CALIBRATION

In a letter dated September 20, 1985, the applicant requested that their planned response checks to control rod movements during heatup and Test Condition 1 be deleted for the LPRM calibration test. This testing verifies that the LPRMs are responding to neutron flux and provides confidence that when the APRMs are calibrated under the constant rate heatup calibration, a representative calibration is obtained.

The staff notes that the requested change is a level of detailed implementation which does not appear in the FSAR test abstract. The staff recognizes that early confirmation of control rod movement and proper installation of the LPRM detectors are valuable when the detector output is in its useful range, and that, until the threshold of usefulness is attained, response checks of this type are of little value. However, the low-power license for the facility will be conditioned to prohibit operation above 5% of full thermal power. The only way to assure this power level is not exceeded is based on the APRM constant rate heatup calibration, which requires proper LPRM calibration. For this reason the staff believes that the LPRM response checks should remain in the test.

3. TEST NO. 10 - AVERAGE POWER RANGE MONITOR CALIBRATION

The objective of the APRM Calibration Startup Test is to calibrate the APRM system with respect to the plant heat balance. During initial heatup, at power levels too low for a heat balance, the first APRM adjustment is accomplished by adjusting the APRM gains so the APRMs indicate slightly

higher than an estimated power level, calculated by using the heatup rate and other plant parameters. By letter dated October 17, 1985, the applicant proposed using preselected gains based on startup experience at other plants to initialize the low power adjustment of the APRM system. The staff notes that this is a level of detailed implementation which does not appear in the FSAR test abstract. It is the staff's belief that the startup experience from other facilities does not necessarily demonstrate that preselected APRM gains will provide conservative indication of reactor power levels. It is the staff's position that these preselected gains should not be used.

The Technical Specifications require surveillance testing for APRM system calibration at conditions where the APRM startup testing calibration is performed. The applicant requested combining the startup test and surveillance test for calibration of the APRM system into a single test to eliminate redundant testing. The objectives of both tests are identical. The staff discussed the combining of surveillance and startup testing in the safety evaluation for Test No. 17, transmitted by letter dated January 22, 1986. As stated in the January 22, 1986 letter, the staff finds the combination of surveillance and startup testing to be acceptable if (a) the test objectives are the same; (b) it is consistent with regulatory guidance, and (c) if combined as discussed in the January 22, 1986 safety evaluation, the same level of review, approval and data evaluation of a startup test procedure is retained.

4. TEST NO. 23A - MSIV FUNCTIONAL TESTS

In a letter dated October 17, 1985, the applicant proposed changing Test No. 23A- MSIV Functional Tests. The proposed test change deletes determining the maximum power condition for MSIV functional testing. The applicant stated that future full closure tests of the MSIVs to satisfy plant technical specification requirements will be done at cold conditions (as is done at Limerick). The testing at cold conditions is conservative because the MSIVs, by their design, close faster during the rated conditions (when steam is flowing in the steamlines). Therefore, the testing to determine the maximum power level for subsequent testing is not required. A one time test of full MSIV closure will be performed. Power Ascension Test No. 23B, MSIV full closure test is scheduled to be performed at full power to satisfy the regulatory requirements. The proposed change to delete the testing of MSIVs to determine the maximum power level for subsequent surveillance testing is acceptable.

5. TEST NO. 5 - CONTROL ROD SCRAM TIME TESTING AT HOT STANDBY WITH FULL REACTOR SCRAM CONTROL ROD TIME TESTING

In a letter dated December 9, 1985, the applicant proposed modifying Test No. 5. Test No. 5 is concerned with testing of control rod drives. As part of the test, rods are scram time tested after fuel loading at cold shut-down. During reactor heatup, four selected rods are time tested at various reactor pressures. Normally, all rods are time tested at rated pressure and low power, and again four rods are selected for testing during power ascension. The applicant has proposed that this sequence be

modified such that the test of all rods at rated pressure and low power be replaced with a test of four rods, and a test at approximately 20 percent power of all withdrawn rods as part of the scram in the Loss of Offsite Power test. This would result in the 37 "Control Cell Core" (CCC) rods not being fully tested since they would not be fully withdrawn.

General Electric has analyzed the reactivity worth of the scram over the insertion range of interest to transient analyses and found only a small difference in reactivity insertion between the scram of all rods and a scram in which the CCC rods and 8 other (assumed) extreme, inoperable rods do not insert. (The CCC rods are low worth rods.) They have also reexamined the transient analyses of events with this reduced scram function and determined a minimum critical power ratio (MCPR) penalty to be applied for the assumption that these rods do not participate in the scram. The operating limit MCPR (OLMCPR) is increased by no more than 0.01 for either the ODYN option A or for the curve of OLMCPR vs measured scram time for ODYN option B. The proposed ODYN scram time would involve only those rods fully measured in the scram time test.

In the December 9, 1985 letter, the applicant proposed Technical Specification changes which would (1) exclude for first cycle only, the 37 CCC rods from scram time tests in Specification 4.1.3.2 and .3 and (2) increase the OLMCPR by 0.01 in Specification 3/4.2.3 and the related figure.

Based on the above information, the staff finds that (1) all rods will have been scram tested at cold conditions, (2) four rods will have been tested during heatup, including full pressure, and thus indicate any departure from normal trends, (3) analysis indicates little reactivity worth change during times important for transient analyses, and transients have been reexamined assuming no scram at all for CCC rods and 8 inoperable rods and suitable OLMCPR adjustments have been made to account for changes, and (4) suitable Technical Specifications have been proposed to exempt CCC rods and to increase OLMCPR. However, the applicant has not addressed the affect the proposed test modification will have on the shutdown margin technical specification. Additionally, performing the first full-core scram at rated conditions (temperature and pressure) during a loss of offsite power test is not acceptable. Because this would be the first scram under operating conditions, there is no assurance that a successful full-core scram will be realized. Additionally, should the scram not occur, the plant would be in an anticipated transient without scram situation. Although the applicant has provided sound technical justification from a core physics perspective why the proposed test modification should be accepted, the above concerns are of such significance that the proposed test modification, in its current form, is unacceptable.

6. TEST No. 23B - MSIV FULL ISOLATION

In a letter dated October 17, 1985, the applicant proposed substitution of an inadvertent MSIV closure for the planned MSIV closure test. The staff finds the proposed substitution of an inadvertent full MSIV closure for the planned full MSIV closure test to be unacceptable for the following

reason. Although precedent exists for the substitution of inadvertent transients for planned tests, the acceptability of the substitution has been determined only after thorough analysis to assure that the proper quantity and quality of data was obtained which demonstrated that the inadvertent transient fulfilled all the objectives of the planned test. The evaluation of an inadvertent full MSIV closure can only be performed after the transient has occurred. Therefore, pre-approval to substitute an inadvertent transient for a planned test is inappropriate, and the proposal is unacceptable.

APPENDIX T

TECHNICAL SPECIFICATION CHANGES BETWEEN LOW-POWER LICENSE ISSUANCE AND FULL-POWER LICENSE ISSUANCE

Since the issuance of the low-power license, the licensee has requested certain changes to the Technical Specifications. These changes and the staffs' conclusions regarding their acceptablity are discussed below. Some of these changes modify license conditions.

(1) Technical Specifications 4.6.5.3.b and 4.7.2.b

In a letter dated June 13, 1986, the licensee requested changes to Technical Specifications 4.6.5.3 and 4.7.2 to state that the filtration, recirculation, and ventilation system (FRVS) and control room emergency filtration system (CREFS) heaters will be on, rather than just operable, for the required surveillance. Surveillance Requirements 4.6.5.3.b and 4.7.2.b currently state:

> At least once per 31 days by initiating, from the control room, flow through the HEPA (high efficiency particulate air) filters and charcoal adsorbers and verifying that the subsystem operates for at least 10 hours with the heaters and humidity control instrumentation OPERABLE.

The licensee requested that these requirements be changed to read:

> At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates for at least 10 hours with the heaters on in order to reduce the buildup of moisture on the carbon adsorbers and HEPA filters.

The primary purpose of the required periodical test operation of engineered safety feature filtration systems (10 hours for every 31 days with heaters on) is to reduce the potential buildup of moisture on the charcoal adsorbers and HEPA filters. The only function of the humidity control instrumentation (humidity controllers) is to activate the heaters when relative humidity in the incoming air to the charcoal adsorber and HEPA filter exceeds 70%. The humidity controllers are channel calibrated every 18 months, as specified separately in Surveillance Requirements 4.6.5.3.e.3 and 4.7.2.e.4. Since the surveillance requirement will necessitate having the heaters operable, the licensee's requested changes are reasonable and consistent with the intent of this Technical Specification section. Therefore, the staff finds the requested changes acceptable.

(2) Technical Specifications 4.6.5.3.e.3 and 4.7.2.e.4

In a letter dated June 13, 1986, the licensee requested changes to these Technical Specifications to clarify the method of verifying humidity control operability and to change the FRVS heater dissipation rating.

The requested changes clarify the surveillance requirements for the humidity controllers by stipulating the performance requirements for humidity controller testing in the last sentences in Technical Specifications 4.6.5.3.e.3 and 4.7.2.e.4 as follows:

> Verifying humidity is maintained less than or equal to 70% relative humidity through the carbon adsorbers by performance of a channel calibration of humidity control instrumentation.

As discussed in Item (1) above, this change requires a channel calibration of the humidity controllers every 18 months, independent of periodic heater operation, and constitutes clarification of one surveillance requirement for the humidity controllers. The staff finds this change acceptable because channel calibration is an acceptable means of verifying the operability of the controllers.

In addition, the licensee requested a change to the heater dissipation rate specified in Technical Specification 4.6.5.3.e.3 from the currently specified 100 ± 5 kW to 100 ± 10 kW, consistent with the requirement of American National Standards Institute Standard ANSI N510-1980, which allows that the heater dissipation rating can be ±10% of the heater output. The staff finds this request also acceptable.

(3) Technical Specification Table 3.3.7.1-1, "Radiation Monitoring Instrumentation"

The current Table 3.3.7.1-1, Action 74 states that, with the offgas pretreatment radiation monitor inoperable, the release(s) into the pathway may continue for up to 30 days provided:

> a. The offgas system is not bypassed, and
> b. The offgas post-treatment radiation monitor is OPERABLE, and
> c. Grab samples are taken at least once per 8 hours and analyzed within the following 4 hours;
>
> Otherwise, be in at least HOT SHUTDOWN within 12 hours.

In a letter dated June 13, 1986, the licensee requested deletion of Provision (b) above, that is, deletion of the requirement that the offgas post-treatment radiation monitor be operable. The Hope Creek offgas system is designed to provide 35 days and 36 hours of delay time for xenon and krypton, respectively, with a total air inleakage of 75 standard cubic feet per minute. The post-treatment monitor indication is, therefore, not for real-time measurement indication. Instead, grab samples taken every 8 hours (analyzed within the following 4 hours) should be adequate to meet the limiting condition for operation in Technical Specification 3.11.2.7. In addition, everything that passes through the offgas post-treatment monitor passes through the downstream north path vent monitor, which is required to be operable by Technical Specification Table 4.11.2.1.2-1. The staff finds the requested change acceptable.

(4) Technical Specification Table 3.6.3-1, "Primary Containment Isolation Valves"

The current Table 3.6.3-1 requires operability testing of the reactor vessel head seal leak detection line excess flow check valve. In a letter dated

June 13, 1986, the licensee proposed Technical Specification change that would add a notation deleting the requirement to operability test excess flow check valve BB-XV-3649.

The reactor vessel head seal leak detection line excess flow check valve is not normally subjected to primary system pressure. If the reactor vessel head seal should fail, this valve could be pressurized to reactor pressure. However, any leakage is restricted at the source (head seal), by the instrument line orifice, and at the pressure instrument itself. This excess flow check valve has a 10 CFR 50, Appendix J, exemption for Type A leakage testing. Therefore, this valve need be not subject to operability testing by Technical Specification. The staff finds the requested change acceptable.

(5) Technical Specification 4.4.6.1.4

In a letter dated June 13, 1986, the licensee requested a change to Technical Specification 4.4.6.1.4 that would reduce the minimum temperature for closure stud tensioning from 79°F to 70°F.

The limiting $RT_{NDT}$ of material in the flange region is 19°F. The revision is permitted under Paragraph G-2222 (c) of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section III. The current Code states that when the flange and adjacent shell region are stressed by the full intended bolt preload and by pressure not exceeding 20% of system hydrostatic test pressure, the minimum metal temperature in the stressed region should not be lower than $RT_{NDT}$ plus the effect of irradiation damage.

The proposed revision of the Technical Specification is in compliance with the ASME Code and, therefore, acceptable.

References

American National Standards Institute, ANSI N510-1980, "Testing of Nuclear Air-Cleaning Systems."

American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, Paragraph G-2222, 1971 Edition, Winter 1973 Addenda.

# APPENDIX U

## PROBABILITY OF MISSILE GENERATION IN GENERAL ELECTRIC NUCLEAR TURBINES

### 1 SUMMARY

The objective of the NRC staff's review of the General Electric Company (GE) report, "Probability of Missile Generation in General Electric Nuclear Turbines" (submitted by the licensee in a letter dated July 11, 1986), was to evaluate and, if appropriate, approve the methods and procedures utilized by the General Electric Company, Large Steam Turbine-Generator Department, to determine specific turbine system inspection and testing intervals for its utility customers.

During the past few years, the staff has recommended a probabilistic approach to determine turbine rotor inspection intervals and turbine control system maintenance and testing frequencies so as to maintain the as-built turbine system integrity. The GE report describes such an approach generically and, to the extent possible, supports it with test and turbine system operating experience data. The staff recognizes that probabilistic analyses based on limited statistical data, especially for a complex system, will include inherent uncertainties. Nevertheless, when the overall approach includes conservative assumptions that overcome the uncertainties, then the ultimate results can be meaningful.

The staff concludes that the methodology described in the GE report is state of the art and is acceptable for use in establishing maintenance and inspection schedules for specific turbine systems.

Applicants or licensees who accept GE's recommendations, based on this report, should confirm their commitment to the staff and provide a description of their specific maintenance and inspection program including a curve (or curves) of missile probability ($P_1$) versus service time for their specific turbine rotors.

### 2 BACKGROUND

Although large steam turbines and their auxiliaries are not safety-related systems as defined by NRC regulations, failures that occur in these turbines can produce large, high-energy missiles. If such missiles were to strike and damage plant safety-related structures, systems, and components, they could render them unavailable to perform their safety function. Consequently, General Design Criterion 4, "Environmental and Missile Design Bases," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," requires, in part, that structures, systems, and components important to safety be appropriately protected against the effects of missiles that might result from such failures. In the past, with regard to construction permit and operating license applications, evaluation of the effects of turbine failure on the public health and safety followed Regulatory Guide (RG) 1.115, "Protection Against Low-Trajectory



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Turbine Missiles," and three essentially independent Standard Review Plan (SRP, NUREG-0800) sections: Sections 10.2, "Turbine Generator," 10.2.3, "Turbine Disk Integrity," and 3.5.1.3, "Turbine Missiles."

According to the NRC guidelines in SRP Section 2.2.3 and RG 1.115, the probability of unacceptable damage from turbine missiles ($P_4$) should be less than or equal to about 1 chance in 10 million per year for an individual plant, that is, $P_4 \leq 10^{-7}$ per year. The probability of unacceptable damage resulting from turbine missiles is generally expressed as the product of (1) the probability of turbine failure resulting in the ejection of turbine disc (or internal structure) fragments through the turbine casing ($P_1$); (2) the probability of ejected missiles perforating intervening barriers and striking safety-related structures, systems, or components ($P_2$); and (3) the probability of struck structures, systems, or components failing to perform their safety function ($P_3$).

In the past, analyses assumed the probability of missile generation ($P_1$) to be approximately $10^{-4}$ per turbine-year, based on the historical failure rate (Bush, 1973, 1978). The strike probability ($P_2$) was estimated on the basis of postulated missile sizes, shapes, and energies and on available plant-specific information such as turbine placement and orientation, number and type of intervening barriers, target geometry, and potential missile trajectories. (See SRP Section 3.5.1.3 for a description of the evaluation procedures previously recommended by the staff.) The damage probability ($P_3$) was generally assumed to be 1.0. The overall probability of unacceptable damage to safety-related systems ($P_4$), which is the sum over all targets of the product of these probabilities, was then evaluated for compliance with the NRC safety objective. This logic places the regulatory emphasis on the strike probability; that is, it necessitates that $P_2$ be made less than or equal to $10^{-3}$ and disregards all the plant-specific factors that determine the actual $P_1$ and its unique time dependency.

Although the calculation of strike probability is not difficult in principle, for the most part being not more than a straightforward ballistics analysis, it presents a problem in practice. The problem stems from the fact that numerous modeling approximations and simplifying assumptions are required to make tractable the incorporation into acceptable models of available data on the (1) properties of missiles, (2) interactions of missiles with barriers and obstacles, (3) trajectories of missiles as they interact with and perforate (or are deflected by) barriers, and (4) identification and location of safety-related targets. The particular approximations and assumptions made tend to have a significant effect on the resulting value of $P_2$. Similarly, a reasonably accurate specification of the damage probability ($P_3$) is not a simple matter because of the difficulty in defining the missile impact energy required to render given safety-related systems unavailable to perform their safety functions and the difficulty in postulating sequences of events that would follow a missile-producing turbine failure.

Operating experience shows that nuclear turbine discs crack (Northern States Power Co., 1981; NUREG/CR-1884), that turbine stop and control valves fail (Burns, 1977; Southern California Edison and San Diego Gas & Electric Co., 1982), and that disc ruptures could result in the generation of high-energy missiles (Kalderon, 1972). Analyses (Burns, 1977; Clark, Seth, and Shaffer, 1981) show that missile generation can be modeled and the probability can be strongly influenced by inservice testing and inspection frequencies.

During the past few years, the results of turbine inspections at operating nuclear facilities indicate that cracking to various degrees has occurred at the inner radius of turbine discs of Westinghouse design. Within this period, a Westinghouse turbine disc failure occurred at one facility owned by the Yankee Atomic Electric Company (NUREG/CR-1884). More recent inspections of GE turbines have also discovered disc keyway cracking (Northern States Power Co., 1981). Stress corrosion has been identified by both manufacturers as the operative cracking mechanism.

In view of operating experience and NRC safety objectives, the NRC staff has shifted emphasis in the reviews of the turbine missile issue from the strike and damage probability ($P_2xP_3$) to the missile generation probability ($P_1$) and, in the process, has attempted to integrate the various aspects of the issue into a single, coherent evaluation.

Through the experience of reviewing various licensing applications, the staff has concluded that $P_2xP_3$ analyses provide only "ball park" or "order-of-magnitude" values. On the basis of simple estimates for a variety of plant layouts, the staff also concludes that the strike and damage probability product ($P_2xP_3$) can be reasonably taken to fall in the characteristic narrow range that is dependent on the gross features of plant layout with respect to turbine generator orientation; that is, (1) for favorably oriented turbine generators, $P_2xP_3$ tends to lie in the range of $10^{-4}$ to $10^{-3}yr^{-1}$ and (2) for unfavorably oriented turbine generators, $P_2xP_3$ tends to lie in the range of $10^{-3}$ to $10^{-2}yr^{-1}$. In addition, detailed analyses such as those discussed in this evaluation show that, depending on the specific combination of material properties, operating environment, and maintenance practices, $P_1$ can have values from $10^{-9}$ to $10^{-1}$ per turbine-year depending on the turbine test and inspection intervals. For these reasons, in the evaluation of $P_4$ ($P_1xP_2xP_3$), the probability of unacceptable damage to safety-related systems from potential turbine missiles, the staff is giving credit for the product of the strike and damage probabilities of $10^{-3}yr^{-1}$ for a favorably oriented turbine and $10^{-2}yr^{-1}$ for an unfavorably oriented turbine, and is discouraging the elaborate calculation of these values.

The staff believes that maintaining an initial small value of $P_1$ through turbine testing and inspection is a reliable means of ensuring that the objectives precluding turbine missiles and unacceptable damage to safety-related structures, systems, and components can be met. It simplifies and improves procedures for evaluating turbine missile risks and ensures that the public health and safety is maintained.

To implement this shift of emphasis, the staff recently has proposed guidelines for total turbine missile generation probabilities (Table U.1) to be used for determining (1) frequencies of turbine disc ultrasonic inservice inspections and (2) maintenance and testing schedules for turbine control and overspeed protection systems. It should be noted that no change in safety criteria is associated with this change in emphasis.

## 3 SCOPE OF REVIEW

There are essentially two modes of turbine disc failure that can result in turbine failure; one resulting from rotor material failure at approximately the rated operating speed, or one resulting from failure of the overspeed protection systems resulting in excessive rotor speeds.



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Failures of turbine discs at or below the design speed, nominally 120% of normal operating speed, can be caused by small flaws or cracks left during fabrication or those that initiate during operation and grow to critical size either by fatigue crack growth, by stress corrosion crack growth, or by a combination of both of those mechanisms. Cracks in the bore or hub region of turbine discs could eventually lead to disc failure.

Failures of turbine discs at the destructive overspeed can result from a failure of the governor and overspeed protection systems consisting of speed sensing and tripping systems and steam valves. If the turbine is out of control, its speed can increase until failure occurs. For unflawed discs, destructive overspeed is reached at about 180 to 190% of the normal operating speed. In general, failures that occur at destructive overspeed are caused by stresses that exceed the materials tensile strength.

If a turbine disc should burst, high-velocity, missile-like fragments may break through the turbine casing, possibly generating secondary missiles. These missiles have a potential for damaging reactor safety systems. Alternately, the disc fragments could be arrested and contained by the turbine itself. Hence, in evaluating the risk associated with turbine disc rupture, it is necessary to determine whether or not missiles external to the casing can be generated by postulated disc ruptures.

This appendix considers the above possibilities and summarizes the review and evaluation of the GE report, which describes GE procedures for estimating (1) the design speed missile generation probability, (2) the destructive overspeed missile generation probability, and (3) the perforation of the turbine casing by turbine disc burst fragments.

## 4 DISCUSSION/EVALUATION

This appendix presents an overview of the methodology in Section 2 of the GE report where three major components of the methodology are considered:

(1) probability of turbine overspeed
(2) wheel burst probability
(3) probability of casing penetration

The probability of a wheel burst and the probability that a wheel fragment will penetrate the casing will depend on the speed at which a wheel bursts. Turbine speed is close to 1800 rpm under normal operating conditions; however, when an abnormal event occurs, such as load rejection and/or failure of the control system to function properly, turbine speed may reach 180 to 190% of the rated speed. The probability of attaining these various turbine overspeed levels, therefore, is a major component of the methodology.

Another major component of the methodology is the probability of a wheel burst at various operating conditions, which are defined by two important parameters: speed and wheel temperature. The primary failure mode of the turbine wheel is assumed to be brittle fracture caused by the presence of a stress corrosion crack in the keyway near the bore of the shrunk-on wheel. The fracture mechanics calculations include variations in the toughness of the wheel material, in the depth of the crack, in the likelihood of crack initiation, in the ability to detect crack sizes during inservice inspections, and in the rate of crack growth during subsequent service.

The third major component of the methodology is the probability of a wheel fragment penetrating the turbine casing, given the wheel burst at a particular speed. The missile penetration probabilities are based on energy methods (Gonea, 1973) and laboratory tests. The variations involved in these calculations lead to a probabilistic estimate of casing penetration as a function of burst speed.

Section 3 of the GE report describes overspeed protection systems. GE nuclear steam turbines are equipped with three speed-sensing devices for defense against turbine overspeed.

(1) Normal overspeed protection is achieved through the control valves, intercept valves, and check valves.

(2) An emergency overspeed protection device is set to close all steam valves if the speed reaches 110 to 111% of the operating level.

(3) A backup overspeed protection device is set to close all steam valves if the speed exceeds the emergency trip setpoint (112%).

Both mechanical hydraulic control (MHC) and electrohydraulic control (EHC) systems are employed. Failure models for MHC and EHC systems are analyzed by a fault tree method, and the probability of attaining a given speed is calculated.

Section 4 considers wheel burst in both brittle and ductile modes. Operating experience shows that the primary failure mode of the turbine wheels is assumed to be brittle fracture resulting from the presence of stress corrosion cracks in the keyway near the bore of the shrunk-on wheel. After ascertaining the fracture toughness property at various depths, calculations are made to determine crack length at a particular time from the initial service. Considerations in the probability analysis are given to variations in the likelihood of crack initiation, in the ability to detect and size cracks during inservice inspections, and in the rate of crack growth during subsequent service.

The statistical distribution is applied to crack initiation and growth behavior data obtained from inservice inspections performed on the majority of wheels of operating GE nuclear low-pressure turbines. The relevant information can be extracted from these statistical distributions to arrive at a given value for any assigned probability or vice versa. Because of various parameters involved, the time to crack initiation varies significantly from wheel to wheel.

Tests on wheels with laboratory-produced stress corrosion cracks and on those retired from service were used to define the ability of ultrasonic testing (UT) methods to detect and size wheel cracks. The GE analysis of the data shows that the crack depth is 0.07 in. larger on the average than the measured value for the wheel hub. The crack initiation distribution is influenced by the oxygen concentration in the steam and the type of locking ring that covers the keyways. The Weilbull distribution was fitted to the observed field data, and the characteristic life, a Weilbull parameter, is approximated. Because of the limitations of UT equipment, undetected cracks might have initiated, and when these undetected cracks are taken into account and combined with the average crack growth rate from these initiation times, the actual distribution of crack depths can be estimated.



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The report synthesizes stress corrosion crack growth with fracture appearance transition temperature (FATT), excess temperature versus $K_{IC}$, and the calculated $K_I$, the stress intensity factor at various operating conditions.

FATT is determined from the test results on retired wheels and other laboratory-generated test data. The FATT value increases with distance from the surface to the interior of the wheel. The prediction of deep-seated FATT values is based on the regression analysis, which takes into account the range of three nickel alloys. The distributions of points about the median lines are normal with a standard deviation of 28°F. The overall standard deviation is 35°F when the cooling rate, ultimate tensile strength, percent carbon, and percent nickel error-distributions are taken into account.

The toughness of the wheel material can be ascertained from toughness curves based on excess temperature (material temperature minus FATT) and the data generated from valid American Society for Testing and Materials specimens. A semilog relation fits the data below 100°F excess temperature. The data are more widely scattered at a lower excess temperature than at the higher values. Here, the natural logarithm of standard deviation is a linear function of excess temperature. A lognormal relation is used for all upper-shelf values utilizing the Rolf-Novak relation for all the GE shrunk-wheel service data.

The stress intensity factor $K_I$ is determined from a relationship involving a crack shape factor, the stress, the crack depth, and the geometry of the part near the crack. The general shape of stress corrosion cracks is assumed to be elliptical. They are quarter-elliptical at corners and semielliptical in the interior. An average aspect ratio of depth of crack (half the minor axis of the ellipse) to the half-length along the surface (half the major axis) is assumed to be about 0.4, on the basis of a study of three wheels that had several stress corrosion cracks. The average crack shape factor for corner cracks in the keyway under the hub was calculated to be 1.85. The average shape factor of 1.71 for semielliptical cracks under the web was calculated. The log standard deviation for both of these factors is taken to be 0.01. The corrosion crack branching factor distribution is derived from test data on retired wheels and other data reported in the literature. The nominal bore stress is assumed to be lognormal with a standard deviation of 0.02. The keyway geometric function is based on the weight function method applied to the results of finite element analyses.

After obtaining the probability of a crack initiating at time t and knowing crack depth "a" at inspection time $t_1$, and using the Weibull distribution for growth rate, the probability of having a crack depth "a" at time $t_1$ regardless of when it initiates is obtained by multiplying these two probabilities and integrating over the range from zero to $t_1$. Multiplying the probability of having a crack depth "a" at time $t_1$ by the probability of not detecting a crack depth "a" and integrating the product from zero to infinity for all possible crack depths, the probability of missing a crack of any depth at time $t_1$ is obtained. Thus, dividing the probability of missing a crack of depth "a" by the probability of missing a crack of any depth at $t_1$ will give the density function of any undetected cracks at time $t_1$. Various combinations of temperatures and locking devices result in a median value of 0.03 in. with a lognormal standard deviation of 0.24. This shows that there is a 50% probability of the undetected crack size being less than 0.03 in.

The probability of cracks existing when no indication is found is computed by dividing the probability of missing any depth crack at $t_1$ by the sum of the probability of missing any depth crack and the probability of no crack existing at time $t_1$. This probability is the same as the probability of missing a crack at inspection time $t_1$. The difference between the true crack initiation distribution and the observed crack initiation distribution is considered as a percentage difference for a given time. Thus, the percentage of cracked wheels is higher for the true crack initiation distribution than for the observed distribution.

After adjusting the field data based on true crack initiation distribution, the Weibull distribution as a function of the temperature parameter, reciprocal to temperature, reactor type, and type of locking ring, showed that the Weibull slope was close to unity and the characteristic growth rate distribution for the third iteration remains indistinguishable from that of the first iteration.

The probability of wheel burst at any time is a function of speed and temperature during a cycle between two refueling outages. The cumulative probability of burst increases in time since the last inspection. The probability of wheel burst at time $t_2$, $P_B(t_2)$, can occur either when the wheel bursts at normal operation $P_{BN}(t_2)$ or it bursts at abnormal operation $P_{BA}(t_2)$. However, burst at abnormal speed will occur only if there is a burst at normal speed. The annual rate of missile generation during normal operation is calculated by multiplying $P_{BN}(t_2)$ by the probability of a missile given a burst at normal speed. The probability of burst for abnormal events is derived from the assumption that a burst will not occur until the cumulative burst probability exceeds the level attained during normal operation. An abnormal event occurs at a given temperature and a given maximum speed. This probability difference is summed up for all temperature levels for this abnormal event. Further, summing up for all abnormal events gives the probability of external missile generation $P_m(t_2)$, which depends on the speed at which the wheel bursts. Hence, the event (missile) must be integrated over the speed ranges for a given temperature. This difference must be multiplied by the probability of speed and temperature occurring, and summed for all temperatures that can occur for the abnormal event. This probability must be again multiplied by the annual probability of an abnormal event occurring and summed for all possible abnormal events. Thus, the probability of a missile resulting from abnormal events is obtained. The final probability $P_1$ is the sum of the probability of a missile resulting from normal and abnormal events.

The second mode of failure is ductile fracture of the wheel during an abnormally high overspeed occurrence. Failure occurs when the average tangential stress across the wheel section exceeds the tensile strength of the material. Since both brittle and ductile modes are statistically independent, the combined probability of failure is expressed as a standardized normal distribution.

Section 5 of the GE report discusses the values of the casing escape probability of each shrunk-on wheel of nuclear turbines manufactured by GE. Earlier analyses assumed that the energy absorption was due to a gross deformation of many components of the low-pressure turbine casing. However, present tests show that the absorption is a local "punching" mechanism. Electric Power Research Institute

full-scale casing penetration tests consisted of accelerating a 120° segment of an actual turbine wheel at 180% speed of the turbine. Test results show that empirical formulas are overly conservative (McHugh, Seaman, and Gupta, 1983). The actual penetration of the missile is only halfway through the wall when a 8300-lb missile at 450 ft/sec strikes the wall. The range of final energy variation (energy remaining after absorption) is based on normal distribution with two sigma limits.

Section 6 of the report gives an overall determination of a wheel burst probability that is a function of time, temperature, and speed. During a typical normal operating cycle, the temperature varies from 50°F at the start (0 speed) to 220°F at full loading (1800 rpm), then to 120°F after the coastdown. The probability of the annual failure rate is calculated for both the normal and abnormal operating conditions. By combining these two probabilities, the probability $P_1$, the generation of an external missile, is derived.

Section 7 of the report discusses typical results of calculations. To provide further insight into the influence of various factors involved in the method, missile probability calculations have been made for a typical GE turbine used with a boiling-water reactor. Two tables summarize the information for each of the 32 wheels used on the turbine. The median value of calculated deep-seated FATT is given for each wheel together with the measured values of surface FATT and tensile strength. The type of locking ring used with the axial key is also noted. Tables also describe the wheel temperature under full-load conditions and the median value of crack growth rate, which is calculated using this design temperature. Another table describes the results of missile probability calculations for each wheel of the low-pressure rotor turbine at various times since the last inspection. On the basis of these calculations, the risk of missile generation for each rotor and the unit can be estimated.

## 5 CONCLUSIONS AND RECOMMENDATIONS

The methodology used in the GE report for the calculation of disc rupture and turbine missile generation probabilities is a straightforward application of probabilistic concepts to variations in surface FATT and deep-seated FATT, overspeed due to load rejection, and/or failure of the control system to function properly. The fracture mechanics calculations include the statistical variations in the toughness of the wheel material, in the depth of the crack, in the likelihood of crack initiation, in the ability to detect and to size cracks during inservice inspections, and in the rate of crack growth during subsequent service. Because of the mixing of surface FATT and the deep-seated FATT values, the overall standard deviation is a larger value than the staff would anticipate. In this way conservatism is introduced at each step. The population of experimental tests and the actual data from the retired wheels are still small, and this results in a large standard deviation, thereby giving a conservative estimate. The staff finds that the GE crack growth equation gives a somewhat lower growth rate than that by another vendor; however, the allowable crack length is only one-half the critical crack length for the determination of an inspection interval. Again it should be emphasized that the upper-shelf value of toughness is code allowable (200 Ksi-in.$^{\frac{1}{2}}$), which is again a conservative value.

To arrive at the final probability of missile generation under normal and abnormal operating conditions, a series of numerical integrations is required

and this may introduce some uncertainty. However, the missile penetration formula used is conservative (Woodfin, 1983) so that disc fragments as heavy as 4600 lb at velocities as great as 300 mph penetrated less than half the thickness of walls at impact velocities that would have produced complete perforation according to other formulas.

The staff has completed its review of the GE report after it met with GE personnel at Schenectady, New York, to resolve some questions. The staff believes that various safety factors or margins used in arriving at the final inspection interval are adequate and the report describes an acceptable method to determine such inspection intervals.

Therefore, the staff concludes that the report may be used in determining the inspection interval for turbine discs in operating and new reactor plants. The inspection interval will vary from plant to plant on the basis of the type of turbine in service and the previous inspection results. Applicants or licensees who wish to reference this report should commit to the turbine inspection intervals determined by GE and should submit a brief summary of how the GE method is used for their specific turbines. The summary should include a plot of missile probability versus inspection interval.

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Table U.1 Turbine system reliability criteria

| | Probability, $yr^{-1}$ | | |
|---|---|---|---|
| | Favorably oriented turbine | Unfavorably oriented turbine | Required licensee action |
| (A) | $P_1 < 10^{-4}$ | $P_1 < 10^{-5}$ | This is the general, minimum reliability requirement for loading the turbine and bringing the system on line. |
| (B) | $10^{-4} < P_1 < 10^{-3}$ | $10^{-5} < P_1 < 10^{-4}$ | If this condition is reached during operation, the turbine may be kept in service until the next scheduled outage, at which time the licensee is to take action to reduce $P_1$ to meet the appropriate A criterion (above) before returning the turbine to service. |
| (C) | $10^{-3} < P_1 < 10^{-2}$ | $10^{-4} < P_1 < 10^{-3}$ | If this condition is reached during operation, the turbine is to be isolated from the steam supply within 60 days, at which time the licensee is to take action to reduce $P_1$ to meet the appropriate A criterion (above) before returning the turbine to service. |
| (D) | $10^{-2} < P_1$ | $10^{-3} < P_1$ | If this condition is reached at any time during operation, the turbine is to be isolated from the steam supply within 6 days, at which time the licensee is to take action to reduce $P_1$ to meet the appropriate A criterion (above) before returning the turbine to service. |

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Supplement No. 6 to the Safety Evaluation Report on the application filed by Public Service Electric and Gas Company on its own behalf as co-owner and as agent for the other co-owner, the Atlantic City Electric Company, for a license to operate Hope Creek Generating Station has been prepared by the Office of Nuclear Reactor Regulation of the U. S. Nuclear Regulatory Commission. The facility is located in Lower Alloways Creek Township in Salem County, New Jersey. This supplement reports the status of certain items that had not been resolved at the time of the publication of the Safety Evaluation Report. This supplement supports the issuance of a full-power license to operate the Hope Creek Generating Station.

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